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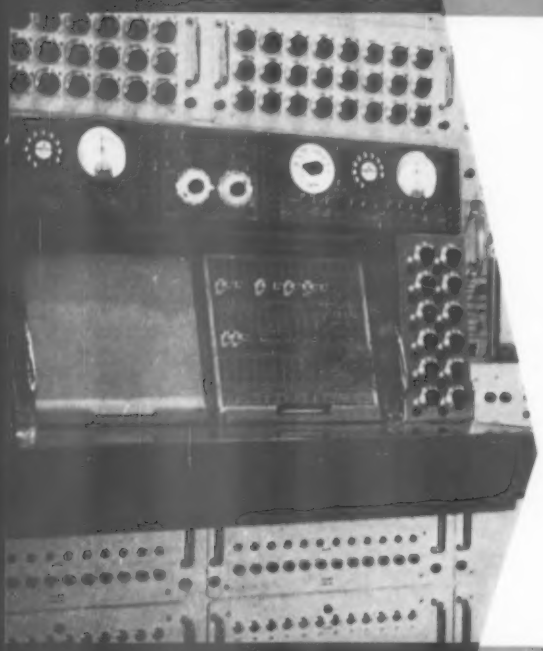

DIGITAL
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Computers in Nuclear Engineering

A Special Report . . . p. 53

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VOLUME 15 · NUMBER 5

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About NUCLEONICS

The distinguished group of gentlemen that you see pictured below are the men behind NUCLEONICS. (There are some women, too, but they were not present when this photograph was taken—at our once-a-year editorial-sales meeting). We are particularly proud of the three new members of our editorial staff shown here for the first time. Taking them in order of date of arrival on the staff, here is a brief rundown on them. Standing fourth from the right, is Assistant Editor Harold L. Davis, freshly arrived from having spent more than two years on aircraft nuclear propulsion work at Pratt & Whitney. With a doctorate in physics from Cornell, Hal becomes the third such on the staff (the engineers will have to fight to keep their heads above water) and will be responsible for the nuclear engineering section of the magazine and for the reactor coverage of NUCLEONICS.

The second new editor is Jesse Cook, standing sixth on the right, who got his engineering background at Annapolis. Before joining the Knolls Atomic Power Laboratory to do technical editing, Jesse worked for the National Bureau of Standards and the Vitro Corp. He will assist Davis on nuclear engineering matters.

The third new man is Robert Otterbourg (standing, second on the left). He is a journalist with experience with Fairchild Publications. He will work primarily on our news staff to help develop our Roundup section.

We plan to do big things with this new editorial strength. Typical is last month's report on fast breeder power reactor. This is something we just couldn't have done two months ago, but with the new editors we felt flush enough to turn one staff member loose for more than a month to work on that story. We expect to be doing a lot more special things as a result of being up to full strength for the first time in a long time. If you have any suggestions for subjects important to the nucleonic field that you think merit a symposium in print, we would be delighted to hear from you. The address is below.

MEN OF NUCLEONICS. Seated left to right: Art Director, Hy Phillips; Advertising Sales Manager, Jim Girdwood; Publisher Howard Mateer; Editor, Jerry Luntz; Market Consultant, Wally Blood; Advertising Representatives, Carl Dysinger and Bruce Winner; and Researcher, Tom Vought. Standing left to right: Buyers' Guide Manager, Jack Quint; Assistant Editor, Bob Otterbourg; Advertising Sales Representatives Warren Gardner, Bill Smith and Dick Alcorn; Promotion Manager, Fred Stewart; Advertising Representatives, Walt Luce, and Mike Miller; Promotion Assistant, Arvid Knudsen; Business Manager, Frank Ward; Advertising Representatives, Jim Hauptli and Phil Hubbard; Assistant Editors, Hobie Ellis and Jesse Cook; News Editor, John Kenton; Assistant Editor, Hal Davis; Associate Editor, Tom Cramer; Managing Editor, Dan Cooper; McGraw-Hill Editorial Director, Ralph Smith. Not present for picture: Roy Thompsen, Assistant Art Director and Jim Caratzas, Production Editor.

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Coming Features in NUCLEONICS

Next Month - Enriched- vs Natural- Uranium Reactors

Fission Plasma Reactors

Later Months - EBWR Design and Construction

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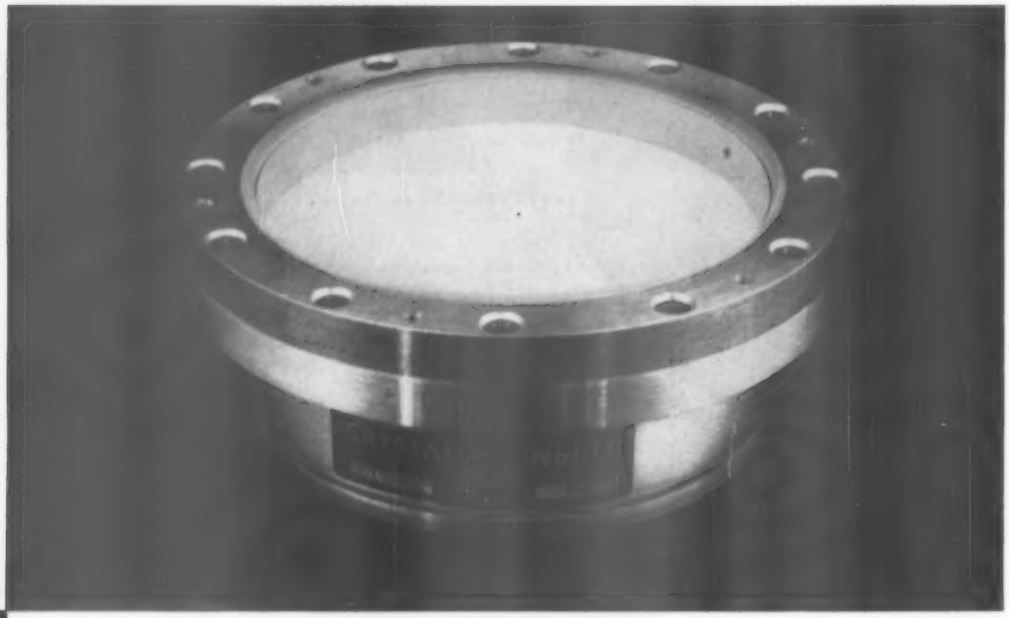

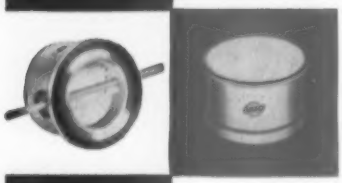

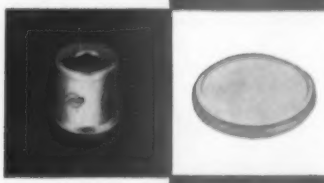

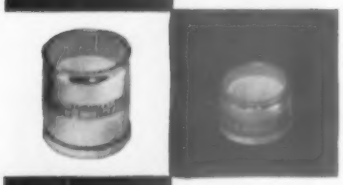

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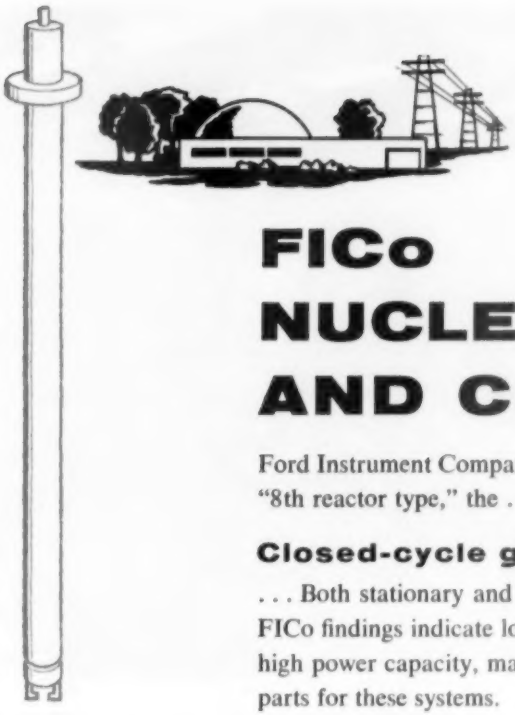

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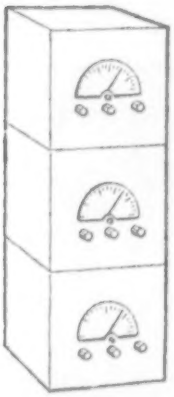

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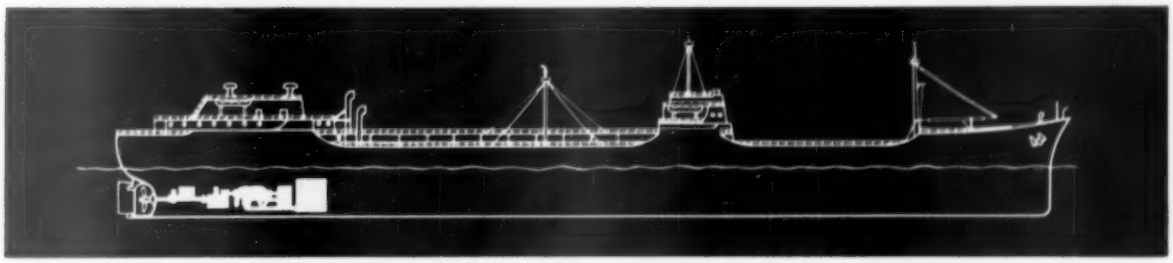

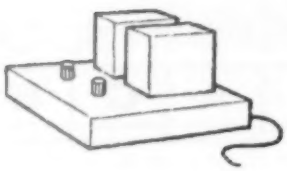

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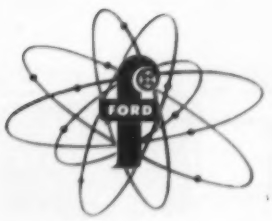

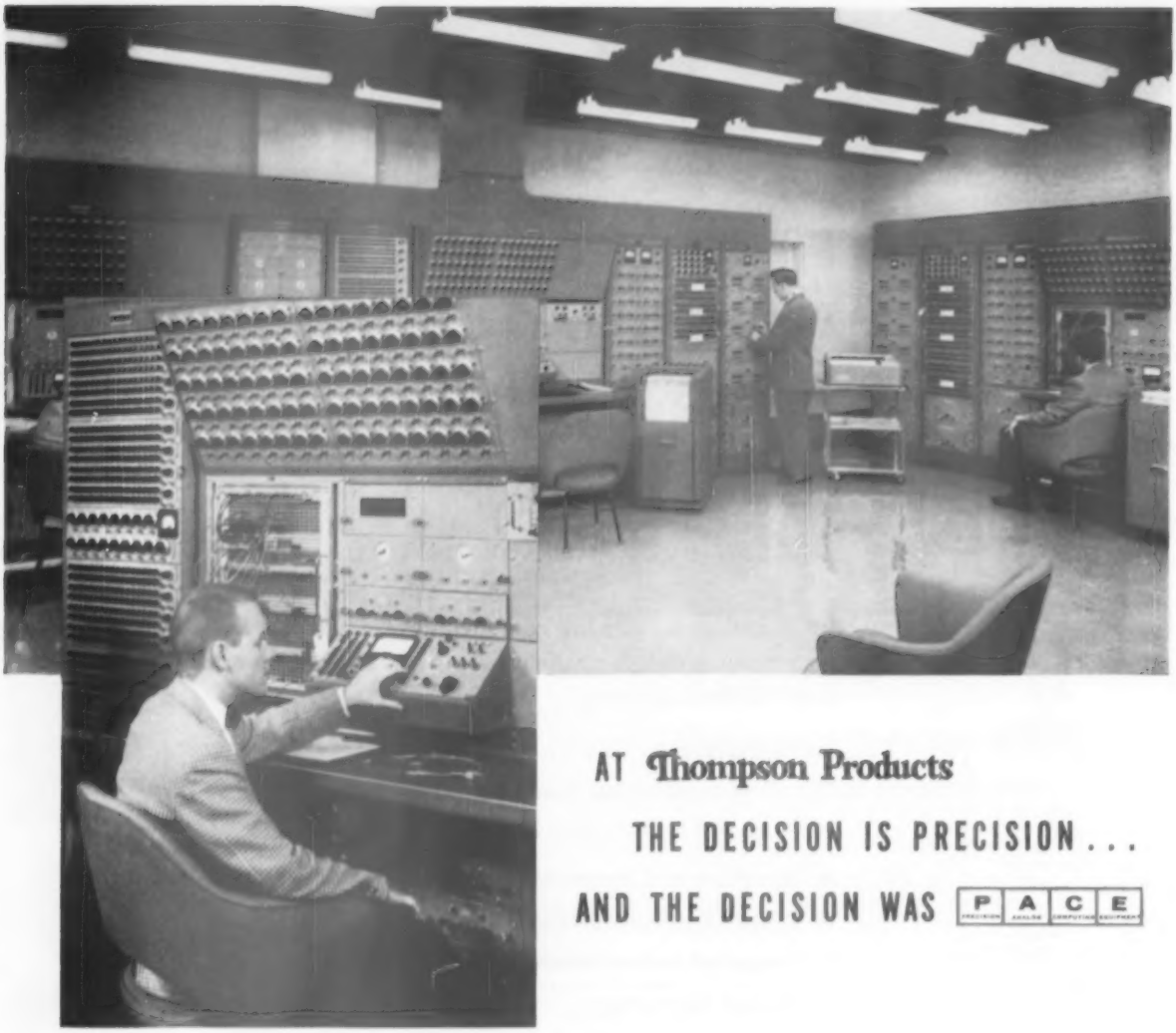

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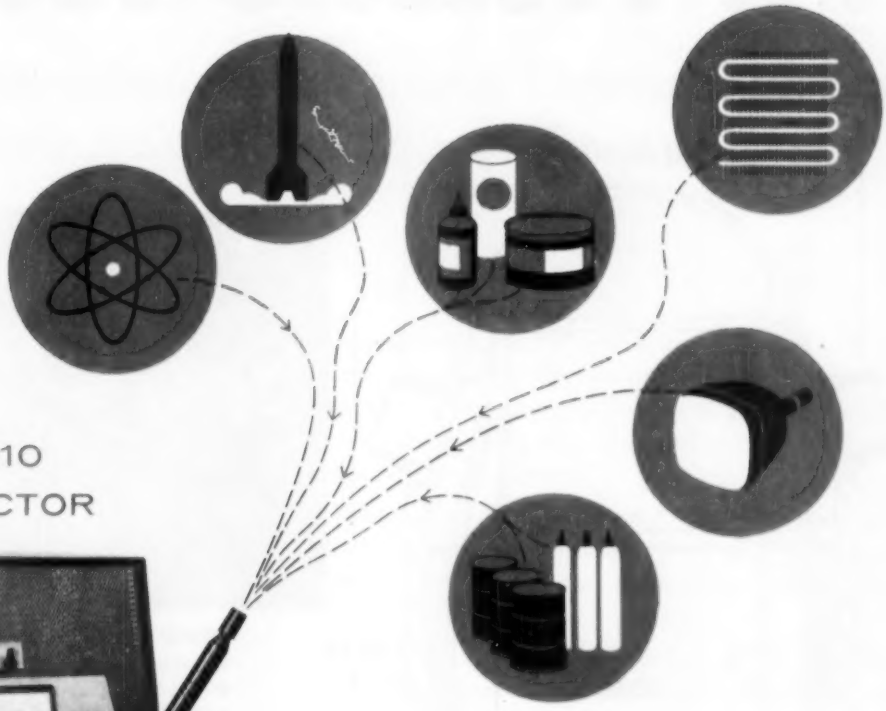

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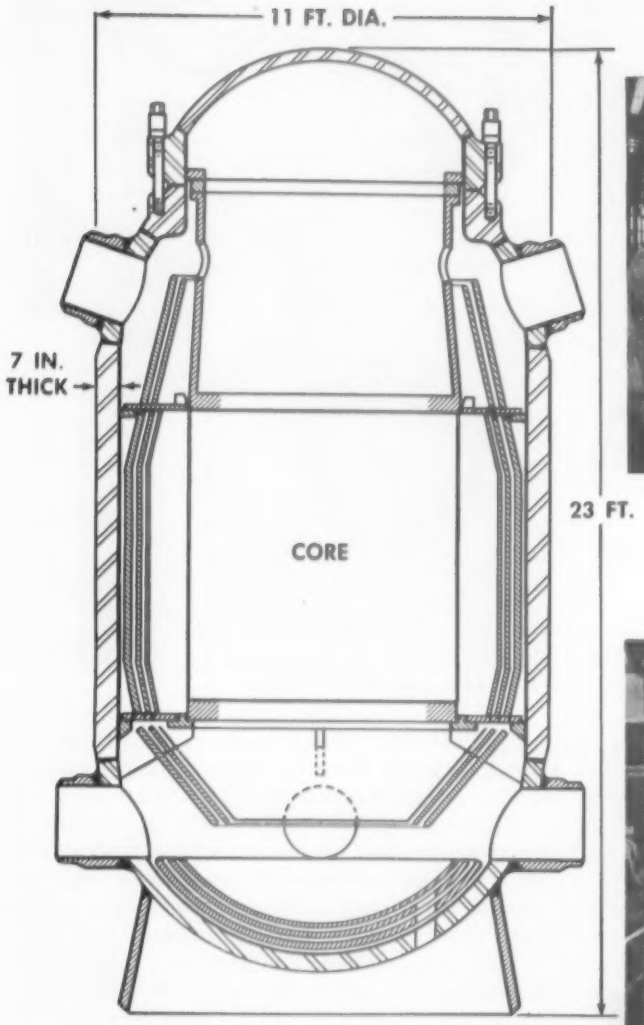

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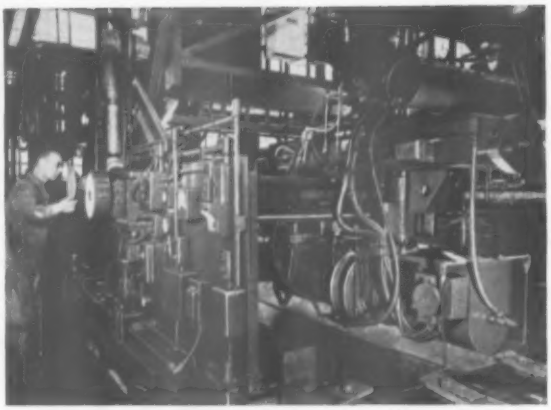

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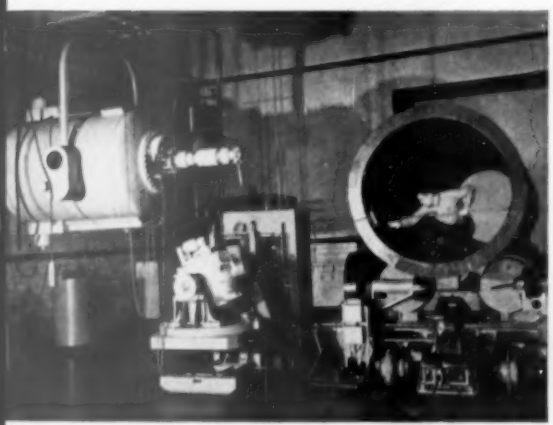

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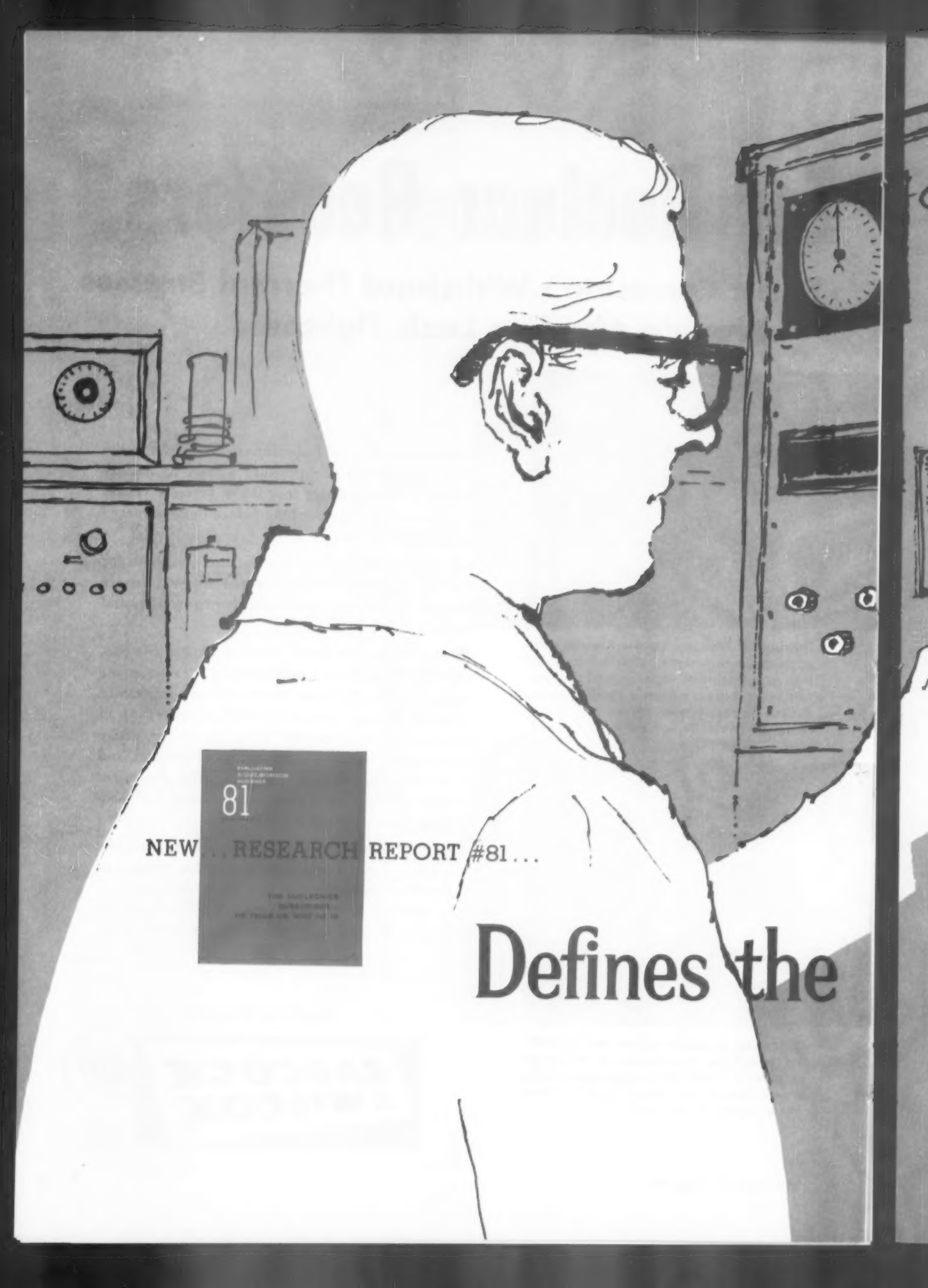
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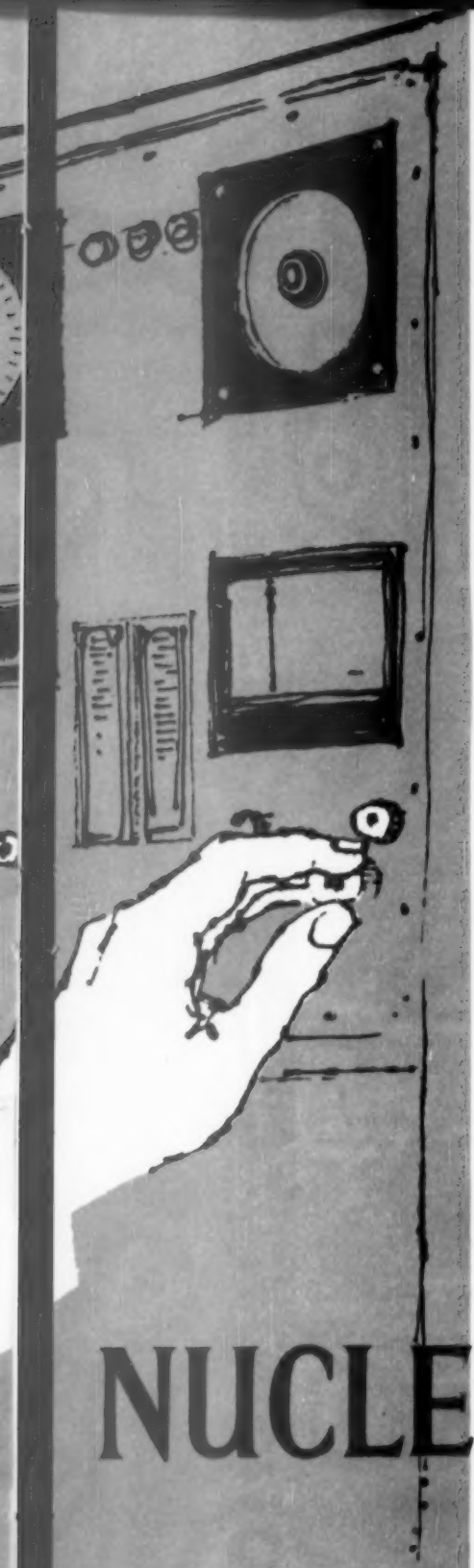

Evaluation Study Shows Why
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. . . in the Growing Nuclear Industry

As part of a continuing effort to keep in personal contact with NUCLEONICS readers, detailed questionnaires were recently mailed to a sample consisting of every fifth subscriber to the magazine.

NUCLEONICS readers responded with a healthy 25% return . . . and here are the *highlights* of several important questions:

What is Your Company's Main Business?
24% of the respondents said, "Research and consulting."
13% said, "Reactor design and construction."
12% said, "Civilian-operated atomic installations."

What is Your Company's Major Nuclear Activity?
25% of the respondents said, "Reactor design and construction."
16% said, "Manufacturer of nuclear components and parts."
14% said, "Research and consultation."
10% said, "Instrument manufacturing."

How Many Fellow Employees Work on Nuclear Projects?
29% of the respondents said, "1-5 people."
11% said, "11-50 people."
10% said, "6-10 people."

Do You Do Consulting Work? What Kind?
37% of the respondents said, "Yes, on radioisotope applications."
18% said, "Yes, on reactors."
18% also said, "Yes, on nuclear components, parts and accessories."

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and his COMPANY

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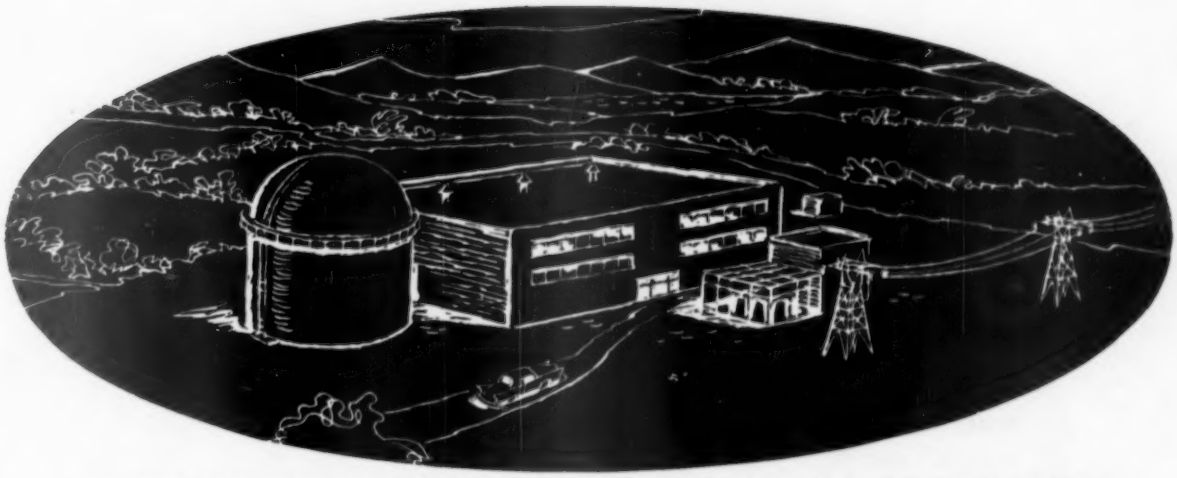

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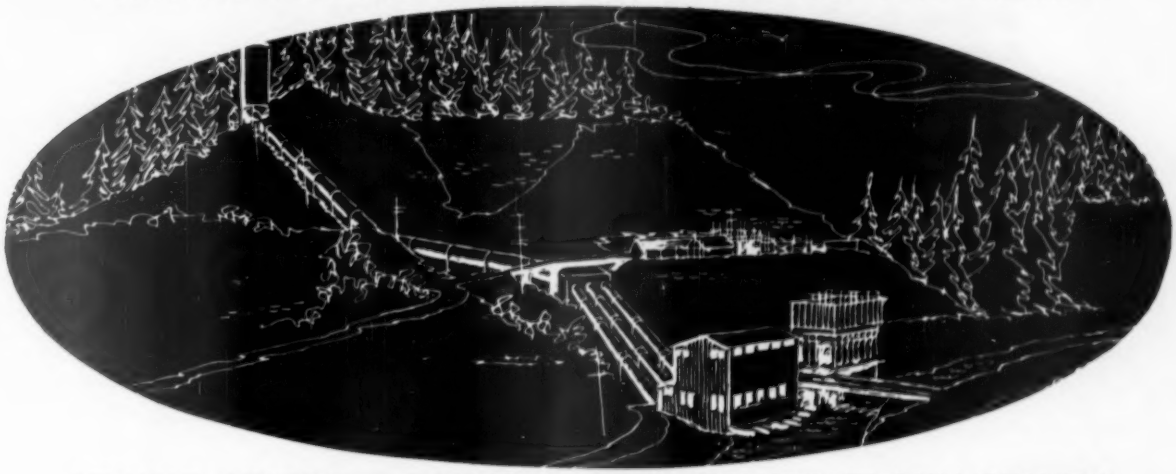

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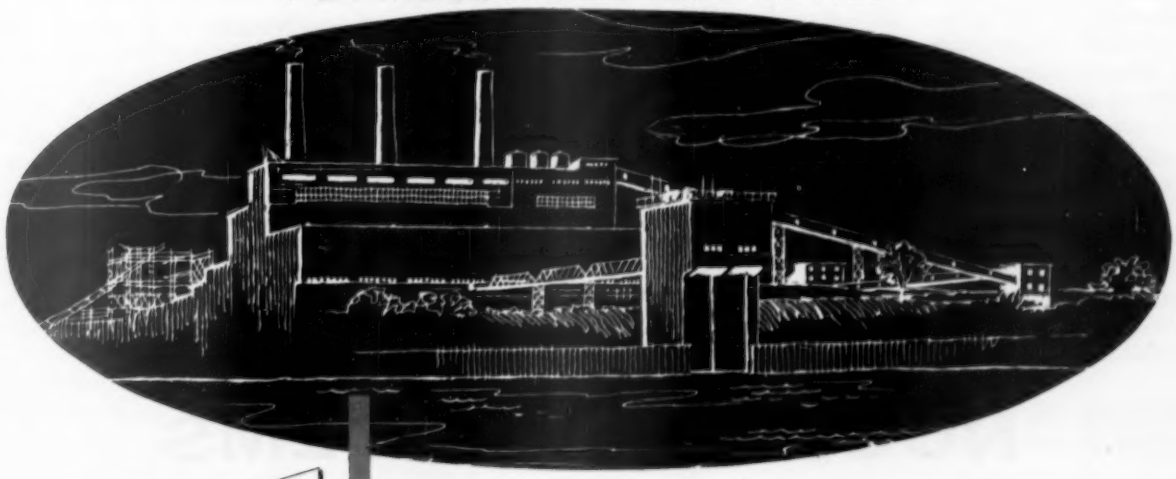

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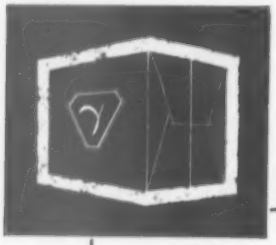

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FOR
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SDR (prototype) is being designed for construction and test operation as a 10,000 KW electric power plant to be installed in the Chugach Electric Association (CEA) system at Anchorage, Alaska, by 1962.

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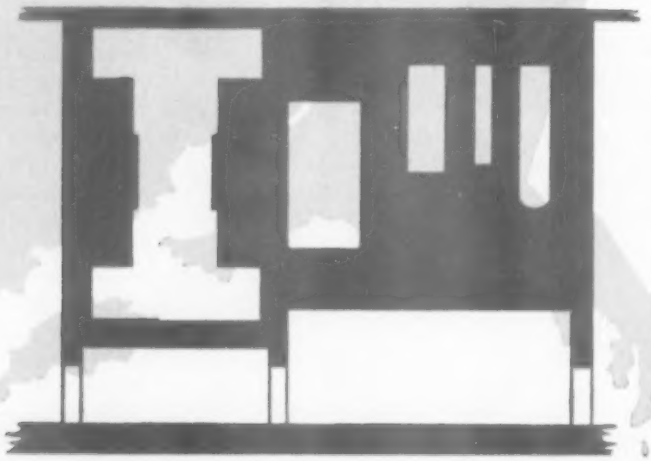
SDR



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White Plains, N. Y.

ROUNDUP of Key Developments in Atomic Energy

AEC Offers Inducements For Private Reprocess Unit

In furtherance of its desire to have private industry provide chemical reprocessing services for private reactors, AEC last month offered to farm out some of its own fuel-reprocessing work to a private firm willing to build a plant, thereby hoping to sweeten the task by increasing and stabilizing the load. Then AEC invited representatives of the chemical industry to discuss terms of a contemplated invitation that might form the basis for a contract covering AEC procurement of reprocessing services from a private firm. Thirty-six companies attended,* some of them non-chemical firms that sat in as observers.

Earlier AEC had announced it stood ready to provide chemical reprocessing services for private reactors pending establishment of commercial reprocessing (NU, March '57, R3; April, 19). But AEC's determined hope is that it get no takers on this offer: rather, that by the time the first generation of private reactors coming on the line in 1960 burn out their first cores, a private facility be ready to handle them. The meeting on April 25 testified to AEC's resolve to end uncertainty about reprocessing and to prod the chemical industry again to enter the field.

AEC had three questions on its mind: if an invitation along the lines outlined were actually made, would the chemical firms submit proposals? If not, what is the main deterrent? and if it be time, when could the firms feel it feasible to bid? What suggestions can be made for alternate bases for proceeding, now or later? It presented the terms of the hypothetical invitation, and described its conceptual plant.

Draft invitation: A major requirement is that the private processing plant would provide capability and capacity—as well as assurance of intention—to handle at reasonable cost (within 15% of AEC's price) spent fuel elements from licensed reactors including those in the Power Demonstration Reactor Program. AEC-owned fuels to be farmed out would include uranium of various U^{235} assays, to be returned as UF_6, and plutonium, to be returned in the form of metal "buttons"; quantities and other details were told to industry, but not the public.

Ten-Year Arrangement

AEC says it would be willing to let "one or more" contracts of this kind for five years with provision for extension for five more years. For evaluating proposals looking to such a contract, AEC would ask detailed breakdown of computation of processing charges that the private plant intended to set, and plant design information including process flow sheet, showing also treatment or recovery of by-product material; plant capacity; starting and completion dates; estimates of major component life under the conditions of operation proposed; development features; proposed method of storage or disposition of fission product wastes and by-product material; proposed detailed operation schedule for the first year of operation.

Conceptual plant description: the conceptual design of MCPP (Multipurpose Chemical Processing Plant) was carried out by AEC "to allow consistent cost estimates to be determined for treating the types and quantities of fuel expected to be available in the period 1960–1967. The cost estimates are needed as a basis for AEC charges for chemical processing of private reactor fuels, in the event private processing is not available." MCPP consists of a number of "head-ends" (handling, mechanical treatment, dissolution and feed storage) operating batch-wise, and a continuous "tail-end" system (liquid-liquid extraction and recovery). Tail-end capacity is one ton/day of normal and slightly-enriched uranium; since criticality is controlled by concentration of solutions, the capacity is reduced for higher enrichments. Thorium treatment rate, because of extractability characteristics, is ½ ton/day, unless it is discarded with the fission-product waste stream, in which case 1 ton/day can be handled. Feed to the plant consists of irradiated fuel elements, which may require limited mechanical treatment prior to head-end dissolution. Products are decontaminated nitrate salts of uranium and plutonium. Head-end equipment is designed to handle the variety of fuels for which reprocessing is expected to be required, and head-end treatment of more than one fuel at a time is possible. Between each batch of fuel treated, a certain length of time is allowed for shut-down, cleanout and startup of the tail-end to prevent mixing of uranium of various enrichments or of feed streams of different chemical composition.

This is the conceptual plant on the basis of whose cost ($20,570,000 to build, $4,592,000/yr to operate) AEC announced in March its $15,300 daily processing charge. (*For additional details see tables p. 22.*)

* American Cyanamid, ACF Industries, Alco Products, AMF Atomics, American Smelting & Refining, Anaconda, Babcock & Wilcox, Baker & Co., Bell Aircraft, Blaw-Knox, Columbia-Southern Chemical, Davison Chemical, Dow Chemical, Electric Storage Battery Co., Ford Instrument, Foster Wheeler, B. F. Goodrich, Goodyear Tire & Rubber, O. G. Kelley, Koppers, Minnesota Mining & Mfg., National Lead, Nuclear Science & Engineering, Glenn L. Martin, Marvin Marcus Consulting, Olin Mathieson, Pennsylvania Salt Mfg. Co., Pickard-Warren-Lowe Associates, Research Chemicals, Inc., Shell Chemical, Sylvania-Corning Nuclear, U. S. Radium, U. S. Rubber, Vitro, Wah Chang Corp., Wyandotte Chemical.

Deadlock Looming in Congress May Stall Atom Work

This country's civilian atomic power program is facing the possibility of being stalled completely as a result of politics in Congress.

Recent political jockeying in Washington has seen the Democrats move themselves into a strong position to bargain for what they want from the Administration and Congress. Republicans are ready to oppose their demands.

All this stems from the continuing clash of philosophies between the two sides—the Democrats wanting a large Federal program of power reactor construction, the Republicans contending that industry with the help of the Atomic Energy Commission is doing the job adequately. But like an argument between man and wife, this one is picking up side issues.

Last year the Democrats lost out when the Gore bill, calling for a $400-million Federal program of reactor construction, was defeated. This year they plan to achieve their objective by hard bargaining.

The Democrats' major objectives this year are adoption of the Gore bill's aims, in whole or in part, and

reappointment of AEC Commissioner Thomas E. Murray, whose term expires June 30. To achieve this they are threatening to:

1. refuse approval of AEC's appropriations requests;
2. refuse to support the Administration-backed Federal indemnity insurance bill;
3. refuse to confirm any nominee by President Eisenhower for the AEC vacancy left by the late Dr. John von Neumann;
4. refuse to ratify the Administration-favored International Atomic Energy Agency statute.

Rep. Clarence Cannon (D-Mo), chairman of the house appropriations committee, has raised the issue of blocking AEC's appropriations. Last month he challenged the legality of the Commission's power program and urged enactment of legislation calling for construction at AEC sites of full-scale atomic electric power plants (the Gore bill).

Cannon Standpoint

Cannon's position is that atomic power program is illegal because specific projects were not authorized by Congress. No further appropriations will be made, he says, unless projects are authorized.

The Congressman says that 80% of the $237-million appropriated or requested for AEC's civilian power reactor program in 1956, 1957, and 1958 lacks specific authorization by Congress; that AEC's "offers of subsidies" under its power demonstration reactor programs contravene the 1954 Atomic Energy Act in several respects; that all four major construction permits issued to date violate the 1954 act; and that AEC's contracts with Yankee Atomic Electric Co. and Power Reactor Development Co. are invalid.

Whether Cannon's interpretation of the 1954 act is right or wrong, he can block AEC's power reactor funds, either within his own committee or on a point of order on the House floor. A ruling by Speaker Rayburn, who is said to be backing Cannon in this move, would block an appropriations vote.

JCAE Democrats can back up Cannon's play. They deny any part in the strategy because, according to Congressional sources, they don't want to be accused of doing anything that might be regarded as delaying the program or of sparking the public -vs- private-power fight. But through the annual bill for authorization of AEC appropriations, they provide specific authorization for various AEC projects; and in providing this authorization, the Democrats could also provide for the construction of large power reactors by AEC, thus achieving the intent of the Gore bill.

Democrats contend that AEC would not be in a good position to argue against this in the case of at least two large reactor types. These are the natural uranium gas-cooled, and the aqueous homogeneous reactors. They claim AEC sought permission from the Budget Bureau to undertake these reactors but was turned down.

Fill 'er up

After steaming 69,138 miles over a period of 26 months, Nautilus was refueled at her home dock for first time (NU, Feb. '57, R5; March '57, R6). Photo shows a little of how it was done: across dock at right is huge floating crane; on Nautilus' deck aft of conning tower is a protective structure built over reactor compartment, connected by canvas passageway to decontamination barge at left. Barge contains complete equipment for monitoring radioactivity levels (note air vent)

The Other Side

JCAE Republicans say the strategy won't succeed. Cannon is wrong in his interpretation of the 1954 act Republicans argue further that the present economy-minded Congress is not going to approve any multi-million dollar Federal atomic power program for which there is little justification.

Murray's reappointment introduces a new factor to the picture. The ten JCAE Democrats plus one Republican—Rep. James T. Patterson (R-Conn.)—in a letter to President Eisenhower have urged that Murray be reappointed. Subsequently, Rep. Chet Holifield (D-Calif.) and House Democratic Leader John W. McCormack, said in effect that without Murray—the last Democrat and Democratic appointee on AEC—the commission would be supine to its chairman Lewis L. Strauss. They argued for retaining someone with "independence of thought" so as to keep the JCAE fully informed of what is going on.

President Eisenhower, his assistant Sherman Adams, and Strauss appear to be or are known thoroughly opposed to retaining Murray.

Because there appears to be no room for compromise on this issue, Murray's reappointment is highly doubtful. Eisenhower can wait until Congress adjourns and then make recess appointments for both Murray and the Von Neumann vacancy and thus delay the fight over confirmation until next year.

Indemnity Involved

Part of the log-rolling situation building up is the Federal indemnity insurance measure which JCAE voted to report out early this month. Sen. Clinton P. Anderson (D.-N. M.) sponsored the bill that would provide $500-million liability coverage for projects in case of major accidents. All sides agree on the need for this legislation, but the Democrats may withdraw their support unless they obtain concessions elsewhere in the atomic program. JCAE Chairman Carl T. Durham (D-N. C.) will push for adoption of the insurance measure.

The bill incorporates another

Anderson measure which requires AEC to hold public hearings on all reactor construction permits and to make public the findings of the Reactor Safeguards Committee. At the same time, the status of this committee would be changed to make it an official body instead of the advisory group it is now.

The insurance issue this year reached a climax when General Electric Co's Vice President Francis K. McCune told Congress his company would withdraw from large atomic power projects unless Federal indemnity protection was voted this session. Other large firms in atomic energy are reportedly willing to accept the Federal Government in the reactor field if that is the price that must be paid for obtaining indemnity coverage.

Compromise?

All the indicators point to the need for compromise if the atomic power program is not to be delayed. One of the likeliest areas for compromise will be on the authorization bill: AEC will be given the authorizations necessary to obtain funds from Cannon's committee, but will have to agree to build some reactors.

Republicans believe they can hold the line on Federal reactor construction because of the economy wave in Congress, and are counting on continued opposition from coal-mining states to the Gore bill.

The outcome is almost certain to be to make AEC more closely accountable to Congress for all its actions.

Full Steam Ahead For Nuclear Navy

• USS *Skate* (SNN 578), the Navy's third A-sub and first of the new "fleet class" boat smaller than the oversize *Nautilus* and *Seawolf*, will be launched May 16 at the yards of General Dynamics' Electric Boat division in Groton, Conn. About the size of the fleet subs of World War II, short and stubby, her keel was laid July 21, 1955—the day *Seawolf* was launched; her pressurized water reactor is being built by Westinghouse. She will be christened by Mrs. Lewis L. Strauss. Eight officers and 75 men—a smaller crew than *Nautilus*'—will man her. Ironically, an earlier sub to bear the same name, after sinking 10 Japanese ships in the Pacific, wound up as a Bikini target ship, survived two blasts, had to be towed to port and scrapped.

• Keel for the Navy's ninth A-sub, *Halibut* (SNN 587), was laid at Mare Island Naval Shipyard near San Francisco. She will be the first nuclear-powered guided-missile sub. Two other A-subs are underway at Mare Island.

• Westinghouse has obtained contracts for propulsion equipment for five A-subs, including steam generators, turbines, gears, and condensers. They will power *Skate* and her three sister subs, and *Halibut*.

• Foster Wheeler and Combustion Engineering will build one each of the two reactor vessels for A1W, the land-based Idaho prototype for the Navy's first nuclear carrier; Foster Wheeler and Westinghouse will build two each of the four steam generators (heat exchangers), NUCLEONICS learns. For the eight-reactor carrier itself, the companies will split the work in the same ratio: FW and CE will each build four reactor vessels, and FW and Westinghouse each 16 steam generators. In addition, Babcock & Wilcox is building the reactor vessel and two steam generators for *Skipjack*, and six reactor vessels for the six other submarines of that class; FW is building 12 steam generators and six pressurizers. For the USS *Long Beach*, the first nuclear cruiser, CE is building the two reactors vessels and FW eight steam generators. CE has the contract for one reactor vessel and two steam generators for the S3G project, the land-based prototype for the *Triton* two-reactor submarine, and the two vessels and four steam generators for the seagoing plant.

Westinghouse gets contract for Belgian Reactor . . .

After ten months of being sidetracked, Westinghouse's project for an 11.5-Mw power reactor for Belgium is back on the road. The company signed a contract with the Société Cooperative Electro-Nucléaire for delivery of the pressurized water, slightly-enriched uranium oxide fueled reactor. Originally when a provisional agreement was announced in November 1955, the reactor was to have been placed in Brussels at the 1958 World's Fair for which it was to supply both electric power and a main attraction, but objection to the site in June 1956 because of its nearness to the Royal Palace forced abandonment of this plan (NU, Feb. '57, R9). Now it is to be built at the Mol nuclear center, for completion by the end of 1959.

. . . and PRDC, Chugach get AEC Contracts

Completion of contract arrangements with Power Reactor Development Co. and with Nuclear Development Corp. of America for power reactors in the AEC's demonstration program were announced by the Commission. PRDC will get $4,450,000 in AEC research and development aid toward the $47-million cost of its Enrico Fermi nuclear power plant (of the total, $33-million is for the 100-Mw fast breeder reactor portion of the plant); and also waiver of fuel use charges through the first five years of the plant's operation, estimated to be worth $5-million. AEC would do the chemical processing, but PRDC bears the cost. The company can cancel the contract if satisfactory liability insurance is not obtainable. The contract with NDA on its proposed 10-Mw sodium-deuterium reactor for Chugach Electronic Assn. of Anchorage, Alaska, covers only the first of three phases—research, development and preliminary design—into which the work was divided (NU, Sept. '56, R1); AEC's contribution to this phase is estimated at $1,725,000. Negotiation of a second contract is underway, covering the remaining phases—final development and design, and construction and operation—which will be done only if the first phase shows the concept to be feasible. The contracts are the second and third to be signed of seven proposals accepted—Yankee Atomic Electric's contract was signed in June (NU, July '56, R10; March '57, R3); still under negotiation are Consumers of Nebraska, Elk River, Wolverine and Piqua.

Euratom Sets Plan Details; Most Feel Goal Feasible

A report on how Euratom might go about meeting its ambitious goal —15-million kw of nuclear power by 1967 (NU, March '57, R2)—has been in preparation in Luxembourg, Euratom capital, by the group's "three wise men" (Louis Armand of France, Franz Etzel of Germany and Francesco Giordani of Italy) aided by U. S. AEC technical experts. The report—due to be published early this month—is designed to answer the jeers of skeptics and to help push the Euratom treaty, signed March 25 in Rome, through the ratification stage. Observers who have felt that the "how" of the Euratom program has been rather vague until now were looking to that report to provide details.

Meanwhile NUCLEONICS polled its correspondents in the capitals of "Little Europe" (the six Euratom nations: France, Germany, Italy, Netherlands, Belgium and Luxembourg) to try to determine whether informed and responsible opinion considered the 15-million kw goal feasible or starry-eyed. Only Germany and Italy reported any skepticism.

German Atomic Minister Siegfried Balke says Germany will be able to install three nuclear power plants of 100 Mw each by 1965, at a cost of $120-million. However he believes not more than 500 Mw total of nuclear power can be completed in Germany by that year. German officials doubt whether the Euratom goals will be reached on time; the feeling is that while the "three wise men" may have a broad philosophical idea of how Euratom should function, the engineers and technicians still have not begun work. With Germany's basic atomic law not yet passed by Parliament, all activities to date have been in the hands of private firms.

Italian scientists of the stature of Prof. Felice Ippolito, secretary general of the National Committee for Nuclear Research (CNRN), stoutly maintain that the goal is "entirely feasible." He is supported by a number of his colleagues, including Giordani (who besides being one of Euratom's three "wise men" is president of Italy's new National Research Center) who are confident that Italy will build its 3-million kw share of the Euratom program by 1962.

But they are in the minority. In the view of all but a handful of Italian experts, it is extremely doubtful whether Euratom can make it; few believe the 15-million kw goal can be achieved or is, in fact, anything more than "a statement of laudable intentions."

Ippolito and Giordani both admit that the Euratom goals can be realized only if the respective parliaments not only ratify the treaty quickly, but also provide the considerable quantities of capital required. U. S. financial and technical support, pledged at the Washington meeting in February, is assumed.

Skepticism is most pronounced in Italian private industrial circles. A number of adverse factors are cited, not least of which is the snail's pace of preparatory work so far. Private interests regard "government interference" as a potentially retarding factor, and if private capital has shown reluctance to enter the power field, it has been partly because of the massive investments and risks entailed but mostly out of fear of nationalization. The trend in Italy is distinctly toward nationalization of atomic power. Only private concern to take the plunge so far has been Edisonvolta of Milan (NU, Jan. '57, R1). Three other groups have been contemplating reactor plants; assuming all four projects are successfully concluded and that each is about the size of Edisonvolta—130 Mw, Italy would have a total of 520,000 kw nuclear by the end of 1960, for an outlay of $240- to $300-million. Consensus is that Italy simply cannot afford the investment required to meet the 3-million kw goal set by Ippolito and Giordani by 1962 or even 1967. The scientists of course reply that "we cannot afford not to make the investment and sacrifices necessary to achieve Euratom's objectives."

Others Optimistic

In the other Euratom countries, the goal is considered to be not at all out of reach, and it's felt that if the Euratom nations are willing to cooperate in a joint effort—and if the U. S. and U. K. can provide a huge amount of men and materials —then the task can be achieved. French leaders give the impression the U. S. has assured them of its willingness to make the effort. A large market for U. S. and British reactors is seen in Euratom nations, some using natural, others enriched uranium.

One warning to American sellers comes from Rome. Potentially Italy is regarded as a good market for American reactors. But Italians are cautious buyers given to shrewd bargaining over quality, price and terms. So far only Westinghouse in the power field has come up with the kind of package that appeals to Italian investors. The British, meanwhile, are reported to be offering "very attractive terms" for Calder-type reactors. American manufacturers are inclined, it is said, to stress cash on the barrelhead and to assume a "take it or leave it" attitude in dealing with potential clients.

Another word of caution from Paris has to do with the Euratom goal generally: the ambitious plans for 15-million kw assume one important political fact: that the Europeans will all pull together, that no "go it alone" demands crop up. They also assume that no other cheaper energy source comes along to take some of the urgency out of the program. Yet the Germans and Italians are both slowly moving into Middle Eastern oilfields, and France claims to have oil in the Sahara.

OEEC Plans Fast Breeder, Homogeneous, MTR Units

Three reactor projects have been approved for joint development and construction by the 17-nation Organization for European Economic Cooperation. OEEC's nuclear experts feel the reactor types having the best chance of finding support among the member countries are an aqueous homogeneous, a fast breeder, and a materials testing reactor. An OEEC study group on experimental reactors, headed by France's Leo Kowarski, was given two months to draw up final plans for joint construction of the homogeneous unit. Sites for the three projects remain to be decided, along with technical details. It is understood the British OEEC delegate has informally invited the agency to locate its first joint venture near one of Britain's research centers. Another OEEC study group, headed by Dr. Pohland of Germany, is drawing up final plans for construction of a pilot chemical reprocessing plant; 14 of the 17 member nations are already supporting this project.

OAS Committee Set Up To Promote Training

A committee on peaceful uses of atomic energy was set up late last month by the Organization of American States; one of its first tasks will be to train nuclear scientists.

Acceleration in France: 2500 Mw by 1965, a Diffusion Plant

Discussions in Paris over the new French five-year atomic energy program, placed before the National Assembly last month, seem to indicate that while France may not have lost its oft-stated faith in plutonium as "the fissionable material of the future," U^{235} nonetheless will play a much bigger role in the French program than had been planned. The new plan, covering the period 1957–61, calls for expenditure of 384-billion francs ($1.1-billion), contrasting sharply with a total of only 150-billion francs ($429-million) spent from the inception of the program in 1945 through the end of 1957. Few details of the program have been revealed as yet, but reportedly involved is a project for construction of an isotope separation plant. This project has stirred up considerable debate and will probably be thrashed out in public when the entire five-year program is debated on the Assembly floor.

Meanwhile Georges Guille, State Secretary for Atomic Affairs has revealed France is now dickering with the U. S. for purchase of more than 2,000 kg U^{235}; the previous purchase contract signed last June was for only 40 kg. He explained that initially the French chose the plutonium route partly because it was less expensive and because France was not in a position to obtain enriched uranium. Now that it has become possible to buy U^{235} from "Anglo-Saxon countries," Guille said, the French AEC "henceforth is able to provide for construction of more and more reactors using enriched uranium"; he added that the only problem now was to determine whether France was to depend on foreign sources for its enriched U. He said a decision had been taken for France to build its own plant, either independently or perhaps under Euratom.

Alternative Plans

Reportedly the French have two projects in mind for construction of an enrichment facility: one plan envisages construction of a large plant costing 60-billion francs ($171-million) but only if Euratom fails to undertake such a project. If Euratom does, then France would build a smaller plant just to be militarily independent.

The five-year program placed before the Assembly notes that by 1965 France should have 800–850 Mw nuclear installed capacity, but it's no secret in government circles that this goal is a bare minimum and that the French AEC and Electricité de France, the national utility, are right now projecting a new goal that assumes ratification of the Euratom treaty. Under this goal it's reported the French expect to hit 2,500 Mw by 1965 and 4,000 Mw by 1967.

Russians Start Operating Largest Accelerator

Successful testing of the Russians' 10-Bev proton synchrotron, world's largest, was announced at the Joint Nuclear Research Institute at Dubna near Moscow. Protons have been accelerated to an energy of 8.3-Bev, and adjustments are being made to achieve 10-Bev.

B&W, George Sharp get merchant ship contracts

A $9,872,000 contract was awarded Babcock & Wilcox for design, development and fabrication of the reactor and components for the propulsion plant to drive the first merchant ship (NU, Nov. '56, R2). The advanced-type pressurized water reactor is to be capable of a developing 20,000 shp continuous output, 22,000 shp maximum, using 2.9%-enriched, stainless-clad fuel elements; it will power both the propulsion machinery and ship's auxiliaries including shipboard electricity supply, heating system and cargo-handling equipment. The B&W contract also covers the steam turbine, reduction gears and turbo-generators. Reactor vessel, shielding, reactor spare parts and other propulsion components including the single shaft and screw are to be contracted for later. George G. Sharp Inc., New York marine design firm, got the cost-plus-fixed-fee contract to design the ship, including preparation of contract plans and specifications, hull design, shielding, ventilation, instrumentation and controls. Preliminary design is to be completed in three months and final contract plans in seven. The naval architectural firm has retained Walter Kidde Nuclear Laboratories for technical assistance in the nuclear phases of the work. The vessel itself, a $40-million joint project of AEC and the Maritime Administration, is to be completed early in 1960. Early in May B&W announced DeLaval Steam Turbine Co. would design and build the ship's main rotating machinery.

AEC formalizes licensing practice, changes inspecting

Another rules change has been proposed by AEC to meet criticism on its methods of handling reactor licensing operations. This would make a part of Commission regulations the practice it has voluntarily been following since the PRDC squabble broke out: to give 15 days' advance public notice of intent to grant a license or construction permit, rather than publishing news of an accomplished fact, as it had been doing prior to the PRDC dispute. An alternative procedure under the proposed new rule is for AEC to direct, on its own initiative, the holding of a hearing prior to issuing a permit or license. The mandatory feature of the amendment would not apply to export licenses. The announcement came a few days after issuance by the Joint Committee on Atomic Energy of a study on AEC's licensing procedures. . . . The PRDC hearing, incidentally, resumes on May 13 when the unions will present their own witnesses, not yet identified. It was learned, however, that the unions' technical advisor is Bernard I. Spinrad, head of reactor engineering at Argonne National Laboratory . . . AEC has established inspection groups in nine Operations Offices throughout the country. Each office has a specific geographic area in which it is responsible for inspection of access permit holders and licensed users of fissionable material and isotopes. However production facility and reactor inspection will continued to be made from Washington.

Waste Disposal, Plutonium Plants Sought by AEC

AEC is asking Congress for $4-million for design and construction at the National Reactor Testing Station of a waste disposal system using the Brookhaven-developed calcination system (NU, Sept. '56, R2). It will provide for calcination of waste streams to dry, solid oxide forms and subsequent storage of the solids, thus reducing the volume of tankage and the hazards of storing the highly-corrosive liquid wastes; by reducing the cost of containment, the facility will aid in reducing reprocessing costs attributable to waste disposal. Another AEC request is for $10-million for a fuels technology center at Argonne that will permit cold plutonium handling in kg quantities and hot plutonium handling to 300,000 curies. Laboratory space for 265 will be provided as well as special-equipment Pu-handling areas. The center is to advance knowledge in use of Pu and its alloys and compounds as nuclear fuel. The plutonium function of the Center will be a new activity in the AEC organization.

Another New ANP Facility; Price Pleased by Progress

Another major new facility for the aircraft nuclear propulsion area at the National Reactor Testing Station in Idaho is underway. Plans and specifications have been issued for a $6–7-million installation for full-scale developmental testing of airplane reactor components during which they can be assembled, tested, dismantled and analyzed from various design standpoints. Included are a large test building, a two-story control and equipment building, miscellaneous supporting buildings, a shielded roadway, four-rail dolly trackage system connecting with other ANP test facilities, contaminated waste system, fuel system, and a chlorination building. Bids are to be opened May 22, the preliminary designs completed by December, and construction in two years. Burns & Roe is working on design.

Meanwhile Rep. Melvin Price (D–Ill.) of the JCAE expressed gratification with progress in the ANP program following testimony by Air Force Secretary Donald A. Quarles. Price and Rep. Durham had criticized the ANP "stretch-out" (NU, March '57, R6; April, 20).

AEC Offers Inducements for Private Reprocessing (p. 17) with these . . .

. . . Estimated Processing Costs for Fuels

	Batch* Size (Kg U)	Processing Rate (Kg U/day)	Plant Time (Days)	Annual Charge ($000)	Unit Charge ($/Kg U)
Fuel Type A	35	28	4.3	65.8	1880
" " "	190	28	13.6	208	1095
Fuel Type B	20†	44	3.5	26.8‡	2680
" " "	60	44	4.4	67.3	1120
Fuel Type C	120,000	1000	128	1459§	12
Fuel Type D	20,000	1000	28	428	21

* Annual, unless otherwise stated
† Every other year
‡ Average for each year
§ Based on $11,400/day processing

Fuel Type A—Plate assemblies of approximately 90% enrichment U as an Al alloy, Al clad; processing composition is 7.5% U, 92.5% Al; assumed burnup is 20% of the U-235.

Fuel Type B—Plate assemblies of approximately 90% enrichment U as a Zr alloy, Zircaloy-2 clad; processing composition is 1.5% U, 98.5% Zircaloy-2; assumed burnup is 25% of the U-235.

Fuel Type C—Slugs and plate assemblies of normal U, Al clad; processing composition is 95% U, 5% Al; assumed irradiation is 800 MWD/T.

Fuel Type D—Rod shaped elements of normal or depleted U, Zr and S.S. clad; processing composition varies but is mostly U or UO_2.

AEC Radiochemical Processing Charges

Reactor	Batch* Size (Kg U)	Processing Rate (Kg U/day)	Plant Time (Days)	Annual Charge ($000)	Unit Charge ($/Kg U)
Commonwealth Edison	14,500	1000	22.5	344	24
Consumer P.P.D., Nebraska	24,400	1000	32.4	496	20
Consolidated Edison	8,140-Th 198-U	1000-Th 24-U	16.1	246	30-Th 1240-U
PRDC					
Core	5,475	150	44.5	681	124
Blanket	32,300	1000	40.3	617	19
Elk River					
Core	12	44	3.3	50.5	4200
Blanket	6,850†	1000	13.8	42.8‡	31
Wolverine	20†	88	3.3	10.1‡	2500
Yankee Atomic Electric	22,000	1000	30.0	459	21

* Annual, unless otherwise stated
† Every fifth year
‡ Average for each year

These charges are for processing delivered fuel elements to decontaminated nitrates, and are based on AEC estimates of various reactors. In practice batch sizes will be determined as outlined in AEC statement of March 12. These are order of magnitude estimates of irradiated fuel processing charges to illustrate pricing policy. They do not constitute a commitment on part of the government.

Fuel Characteristics Assumed:

Comm. Ed.—UO_2 pellets in Zircaloy-2 tubular assemblies; U of 1.5% enrichment; processing composition is 76% UO_2, 24% Zircaloy-2; assumed burnup is about 60% of the U-235.

CPPD—Slug assemblies of uranium of 2.3% enrichment, S.S. clad; processing composition is 91% U, 9% S.S.; NAK bond; assumed burnup is about 16% of the U-235.

Con. Ed.—Alternate U and Th plates in assembly; the U is of about 90% enrichment as a Zr alloy and is clad with Zircaloy-2 as is the Th; processing composition is 53% Zr, 3% U, 44% Th; assumed burnup is 57% of the U-235 and 49% of the U-233.

PRDC

Core—Pin assemblies of U of 27% enrichment as a Mo alloy, Zr clad; processing composition is 72% U, 8% Mo, 3% Zr, 17% S.S.; assumed burnup is about 4% of the U-235.

Blanket—Rod assemblies of depleted U, S.S. clad; processing composition is 80% U, 20% S.S.; Na bond; assumed irradiation is to about 4000 g/t Pu.

RCPA

Core—Plate assemblies of U of about 90% enrichment as a Zr alloy, Zr clad; processing composition is 3% U, 97% Zr; assumed burnup is 40% of the U-235.

Blanket—Plate assemblies of normal U as a Zr-Nb alloy, Zr clad; processing composition is 65% U, 34% Zr, 1% Nb; assumed irradiation is to about 4000 g/t Pu.

Wolverine—UO_2SO_4 solution; U of about 90% enrichment; assumed burnup to 48% U-235.

Yankee—UO_2 pellets in S.S. tubular assemblies; U of 2.5% enrichment; processing composition is 85% UO_2, 15% S.S.; assumed irradiation is 6400 MWD/T.

Report from Puerto Rico:

AEC Spanish-language Training Center to be Focus For All Latin America; Power Reactor Contemplated

SAN JUAN

It is perhaps singularly appropriate that Puerto Rico should be striving to become Latin America's capital of education and training in nuclear science. For this lush, scenic island—once known primarily as a depressed land of a one-crop economy, overpopulation and unemployment, and more recently associated with Operation Bootstrap, establishment of 405 new factories in 12 years, and a diversified, dynamic economy—is also that rarest of things: a harmonious blending of United States and Old World ways of life. The new Commonwealth is the subject of one of the most attractive and promising projects in the U. S. Atoms-for-Peace program, by the terms of which atomic energy would be a means simultaneously of serving directly the expanding needs of Puerto Rico; indirectly, the other nations of Latin America; and in the process serving as a world showcase demonstrating the benefits of atomic development applied through functioning of the democratic process.

The project proper pertains principally to the plans to establish both reactor engineering and isotope-handling courses at the University of Puerto Rico. But a second front, so to speak, has been opened up with the decision of the island's power utility to work with AEC toward construction at once of a 15–20 Mw power reactor for its grid that would be completed in 1960. Both programs upon authorization by Congress would be administered by AEC's Oak Ridge Operations Office.

I—Education

AEC is venturing into a new field —nuclear training in a foreign language—with its project to set up a Nuclear Center at the University of Puerto Rico. This is eventually to be an identifiable AEC facility, which the University will operate for AEC on a reimbursable contract, just as, for example, the University of Chicago operates Argonne for AEC. The Commission has already awarded a grant of $216,950 to the University of Puerto Rico for acquisition of laboratory equipment to support establishment of its nuclear science and engineering curricula (NU, April '57, 132).

The 54-year-old University of Puerto Rico is a U. S. land-grant college with high academic standards, accredited by mainland educational agencies, and with opportunities of scientific training equal to any in the Spanish-speaking world. Instruction is normally conducted in Spanish, although students are bilingual. UPR has a full-time faculty of 960 and 15,000 graduate and undergraduate students—of whom some 10,000 are at the main campus at Rio Piedras, a suburb of San Juan, and 2,400 at the campus at Mayagüez where the engineering and agricultural schools are located. The Nuclear Center will initially embrace two programs, one at each of these two campuses.

Isotopes

At Rio Piedras will be located the Radioisotope Techniques Training Center, which actually already held its first course last month. This was a pilot two-week offering of the isotopes course, given to a group of some twelve faculty members from the University's School of Medicine, the Agricultural Experiment Station, the Colleges of Natural Science and of Engineering—not for the purpose of instruction as much as to give the "students" a broad view of what the Isotope Center plans to do, and to permit them to report back to their various divisions on what is being done. Regular courses in the future will be of four weeks' duration, patterned after the isotope techniques courses at the Oak Ridge Institute of Nuclear Studies (UPR is one of the 35 sponsoring universities of ORINS). Two leading ORINS scientists were at Rio Piedras to conduct last month's pilot course: Lawrence K. Akers covered physics the first week, and Ralph T. Overman chemistry the second. Under the supervision of Amador Cobas, director of the Isotope Center, a counting room with scalers and a sample preparation room have already been fitted out, and a radiochemistry lab is now being installed.

Sometime in June Prof. Cobas hopes to offer the first regular four-week course. This will be limited to twelve Puerto Rican students while the Isotope Center is getting its feet on the ground. In August or September when the second course is slated to be given, 24 students—full capacity of the physical plant—will be admitted, and plans are that about half will be Latin American students from outside Puerto Rico.

Eventually Rio Piedras plans also to provide training in other fields of nuclear science besides isotopes: nuclear physics, reactor physics, and applications of reactors to agriculture, medicine and industry. Reactor design and nuclear engineering, however, are the domain of the engineering school at Mayagüez.

A research reactor, probably a pool-type, is part of the plan for development of the nuclear training center at Rio Piedras. The spacious, palm-tree-dotted campus and nearby Experiment Station here, although only three miles from downtown San Juan, still have good-sized tracts of unoccupied land. AEC has said that if it gets the Congressional green light, it will start construction of the research reactor here next year, together with a research and training building "to serve as a nucleus of a full-scale nuclear center covering physics, chemistry, biology, medicine, and agriculture." Some basic research will be carried out to insure it being a strong training center.

Nuclear Engineering

The second main branch of this two-pronged program is located on another sylvan campus at Mayagüez, the island's third city, 108 road miles or half an hour's flight from San Juan. Here, where a $1.2-million engineering laboratory now under construction represents only a small part of the expansion of the University's facilities, 300 engineers a year will be graduating the five-year course by 1960, as compared with 110 at present. Here preparations are underway for the first class of 24 students who this summer will begin the one-year, senior-graduate level, nuclear engineering course. In time, Mayagüez hopes to have its own training reactor on its campus.

Planning

A symposium, tenth of a series of Oak Ridge regional symposia, was held at Rio Piedras and Mayagüez in January, at which mainland and island scientists and officials explored the possibilities and potentials of the Puerto Rican Nuclear Center (NU, Nov. '56, R12; Feb. '57, R12). But UPR had already been working for a year preparing its plans for submission to AEC, Jaime Benítez, Chancellor of the University, told NUCLEONICS.

"The basic purpose," he says, "is to provide an institutional location for a program which will permit stu-

dents from Puerto Rico and elsewhere to come together at this frontier of knowledge and frontier of cultures. The new scientific vistas which nuclear energy provides may afford a vantage point from whence to look at the basic sciences and to explore new possibilities, and perhaps achieve new knowledge and integration."

Although Puerto Rico is a U. S. Commonwealth and UPR a U. S. college, the island is culturally, linguistically and geographically Latin American, and thus provides a unique focus for an educational facility for the U. S. hemisphere neighbors. Furthermore, Puerto Rico is a good testing ground for nuclear power applications as it has no fossil fuel resources and further hydro development potential is about exhausted. Its industrialization program imposes constantly-increasing power requirements for which nuclear energy may ultimately provide a substantial share. Industrialization also implies the possibility of many immediate applications of radioisotope techniques in industrial processes. Active programs of agricultural and health improvement offer a framework in which such applications can be developed as quickly here as anywhere in the world.

To U. S. foreign policy aims, a Puerto Rican Nuclear Center offers opportunities both to aid the twenty Latin American republics by training their nuclear scientists, engineers and technicians, as well as to improve still further the position of Puerto Rico as the display window of economic and social progress.

To AEC, it would offer an opportunity to provide experience for AEC personnel in handling the types of problems that they will encounter in other international programs.

II—Power

While the academic world in Puerto Rico prepares its Nuclear Center, the Puerto Rico Water Resources Authority is losing no time making arrangements to bring nuclear power into its system.

WRA, which now supplies virtually all the power for Puerto Rico, began in 1908 when its earliest predecessor to provide an irrigation system for the drier south coast. Today it produces more than 99% of the electric power consumed in the Commonwealth, distributes it over a centrally-controlled, 5,530-mi grid to 300,000 customers. With its 20 hydro plants having a nameplate rating of 109 Mw and four steam plants with 202 Mw, WRA generated a total of 1.14-billion kwh in fiscal '56. The Authority, administered and staffed almost entirely by Puerto Ricans, appears as modern, progressive and efficient as any mainland utility, and its proud boast is that industrial development and Operation Bootstrap have not been held back for lack of power—indeed, that on the contrary power availability has been a contributing factor to the success of the development program. As a matter of fact, WRA is one of world's fastest-growing electric utilities: 18–20% growth per year.

Informal exploratory discussions held to date between WRA and AEC regarding plans for a demonstration and training power reactor have been described by both sides as highly promising, and a formal meeting is to be held this month in Washington. Prior to that meeting details of the arrangements are not fixed, but the general plans can be described.

Present Scheme

These contemplate that AEC would contract with WRA for construction and operation of a 15–20-Mw power reactor which WRA would synchronize with its own system. For AEC, the reactor would be an adjunct, though not an integral part, of the academic program in nuclear science: it would be a demonstration of a power reactor in normal operation as part of a regular generating system outside the U. S., and in addition it could be used to train operators. For WRA, there are the same favoring considerations as those that confront mainland utilities: since both fuel availability and economics dictate wide use of nuclear power some time in the future, the sooner its techniques are learned the better. Puerto Rico is a fairly high fuel cost area where the economics will look attractive much sooner. But over and above these arguments there are obvious blandishments in nuclear fuel's high concentration of energy per volume to an island dependent for most of its power (especially in the dry season) on oil shipped in from Venezuela—a supply that could be cut off in a day by a mere transport strike, let alone a big catastrophe.

Total cost of the project, including some development work, land, expert advice, interest, etc., has had a $10-million ceiling placed on it, and there is before Congress for approval at the present session a budget item covering a $9-million contribution by AEC. The remaining $1-million would be put up by WRA. For a country where capital is as scarce as it is in Puerto Rico, $1-million is a quite sizeable amount, WRA Director S. L. Descartes pointed out to NUCLEONICS, but WRA feels it justified in view of the knowledge to be gained. WRA would have an option eventually to acquire the reactor.

Natural Containment

The reactor would be located adjacent to the existing Mayagüez steam plant, which is wedged on a narrow shelf of land between a line of rock hills and the Bay of Mayagüez, at the north end of the city. The proposal is to take advantage of this site to test the Norwegian-Swedish practice of containing reactors underground, by burying the reactor into the hillside facing the bay. Whether this is actually done will depend on the cost of excavating as against the cost of more usual containment.

No reactor type is given preference yet. WRA has already asked Walter Zinn and his General Nuclear Engineering Co. to take a preliminary look at the problem, but there has not yet been time for an analysis of the economics of various reactor types in the Puerto Rican framework. Moreover, AEC counsels giving free rein to industry in the bid invitations to propose plants that the bidding firms have faith in and are willing to back, rather than laying down too narrow specifications.

The schedule—provided AEC and WRA agree on a contract, and Congress votes the funds—calls for invitation of bids Oct. 1, 1957, due April 1, 1958; two months to evaluate the bids, one month for contract negotiations with the winning firm, and signing of a contract set for June 30, 1958. Thirty months are then allowed for design and construction, and plant completion is set for Dec. 30, 1960.

"This is a very tight schedule," comments Descartes, "but we're going to break our necks trying to meet it. For us this is an adventure in the acquisition of a new technique."

Need for A-Power Due In Brazil in '75, Study Says

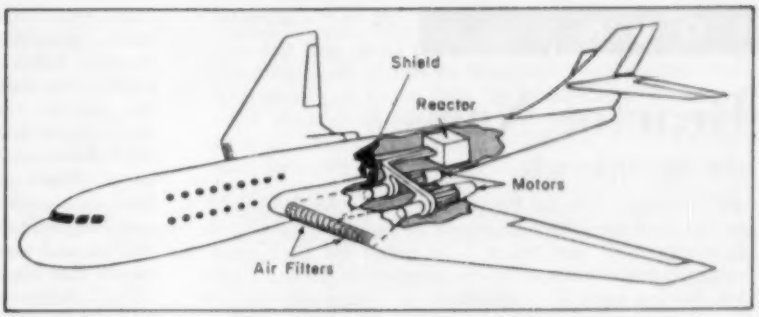
The economic significance of nuclear power to Brazil will be greatest in about two decades, according to a study on nuclear power and economic development in that country just completed by the National Planning Association. Second of a series of country-studies on the productive uses of atomic energy (Japan last June was the first; Israel, India, Italy, Pakistan, Puerto Rico are expected to be published shortly), this report was done in Brazil by Stefan H. Robock.

Hydro Can Fill Bill . . .

The study points out that Brazil has a large undeveloped hydro potential that can be the major source for satisfying energy needs for the next decade or two. Many advantageous sites still exist, and the competitive threshold established for the next decade by hydro projects would be a generation and transmission cost of 12 mills/kwh at 50% plant utilization factor, implying capital investment of $300/kw. With foreign exchange requirements a major factor in Brazil, this means that, except on the seaboard, fossil fuel plants cannot compete with hydro. Nuclear plants, the report says, will in 1965 meet the power costs for thermal plants on the seaboard—11 mills/kwh at 80% plant factor for a large central station plant. In the short run, nuclear fuels will have to be imported into Brazil and will be a drain on foreign exchange resources. To be ultimately self-sufficient in nuclear fuel, Brazil will need uranium as well as thorium minerals and a breeder program or other method of obtaining fissionable material, together with fuel element fabricating and reprocessing facilities.

. . . till A-Power Comes In

From 1965 to 1975, nuclear power prospects should improve steadily, the report says, because of increasing costs for new hydro projects and technological progress in nuclear power. By 1975, reactors may become the principal means of expanding power capacity in Brazil. Judging from existing energy resources, the report concludes, Brazil does not have an urgent need for nuclear power, and could postpone for a few years commitment of large resources in a nuclear energy program, meanwhile accelerating minerals exploration and processing, and technical training.



Nuclear flight, Soviet conception

Drawing of "first nuclear-powered aircraft" from a Russian magazine. The Tupolev aircraft construction team is working on design. Tupolev favors a propulsion reactor heating up charged air sufficiently to create thrust, rather than using reactor heat to drive a steam turbine. Difficulties so far are said to lie in required high temperature for the air passages: 1,650–1,850° F, in turn requiring use of special steels and alloys that are too heavy and have unfavorable neutron cross-sections. However it's pointed out a nuclear plane would fly from Moscow to Vladivostok at 500 mph on 70–75 gm fuel, whereas a Tu-104 (modern two-jet transport) would burn up 150,000 kg jet fuel on same trip. "Certain parts and elements of the projected nuclear aircraft have exceeded the stage of projecting and blueprinting, constructing and testing," writes engineer Y. Sushkov in Komsomolskaya Pravda, the youth journal. "The next stage would be the building of test aircraft first for use on the ground, and later for use as flying laboratories. The first tests with nuclear-powered aircraft are reported to have taken place in the USSR during December 1955. These tests centered around the reactor radiation on the other building materials of the aircraft, and the problem of how to shield the radiation from the other parts of the craft." Whether Sushkov, a member of Tupolev design team, was oversimplifying subject for youthful readers, or whether the few technical details are based on actual Russian work, was not indicated

Harwell: Nero in Service, Zephyr Altered

A new low-energy research reactor at Harwell of the graphite-pile type, Nero can be restacked in a few days; the pile can be varied in size up to a cylinder 14 ft dia × 10 ft high within its 2-ft concrete shield. The new experimental tool first went critical Feb. 23, is being used to investigate physics design problems associated with the more advanced types of graphite-moderated power reactors. Fuel elements of different types may be arranged in numerous lattice patterns. Maximum thermal flux is 10^8; heat output is <100 watts. First assignments: to check calculations on effect of separating fissionable content of fuel, and on a sodium graphite design study.

Zephyr, meanwhile, has undergone a transformation: Britain's first fast reactor, a zero-power assembly, is found to behave like a thermal reactor when graphite replaced uranium in the outer reflector. This slows down the neutrons reflected back into the inner uranium reflector; moreover, the unit can then be controlled like a thermal reactor, by inserting cadmium rods into the graphite. Zephyr thus acts like a half-fast—or slow-fast—reactor on the order of Argonne's ZPR-V (NU, Dec. '56, R7). A chain reaction was obtained with a Pu critical mass nearly same as that used formerly.

AEC Licensing News

Notice of proposed issuance of construction permit: Glenn L. Martin Co., for a nuclear experiment facility at Middle River, Md., containing two independent critical experiment cells, to be completed in June or July, 4/24/57

Export licenses issued: Foster Wheeler, for a 5-Mw tank-type research reactor to Danish AEC, Copenhagen, 4/4/57

Loretz & Co., Los Angeles, for a 500-watt Atomics International solution-type research reactor to Danish AEC, Copenhagen, 4/4/57

Applications received: U. S. Naval Hospital, Bethesda, Md., for AGN-201 research reactor, serial 105, 3/27/57

U. of Virginia, Charlottesville, to build and operate a 1-Mw pool-type research reactor, 3/29/57

Intercontinental Chemical Corp., to export an Atomic International L-54 50-kw solution-type research reactor to Farbwerke Höchst, Frankfurt, 4/1/57

James Loudon & Co., Los Angeles, to export a 50-watt solution-type Atomics International research reactor to West Berlin, 3/28/57

Reactor News

SRE GOES CRITICAL

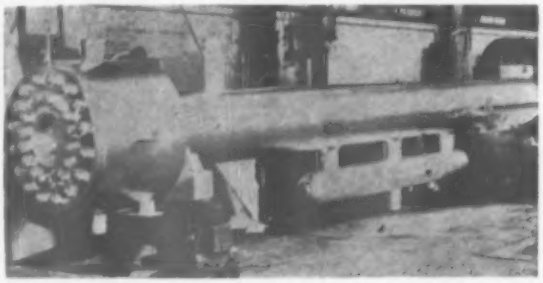
AEC's Sodium Reactor Experiment went critical April 25 on the 33rd fuel element, exactly as had been calculated in advance. It was run at 1 kw power for 10–15 minutes, and has since been taken to criticality several times for varying periods. Calibration of instruments and a complete, detailed checkout of the core and the cooling systems—primary, secondary and auxiliary—are proceeding; electricity will not be generated for some months. SRE, built by Atomics International division of North American Aviation, is second of AEC's five-year program reactors to make neutrons (NU, Dec. '56, R1; March '57, R8).

APPR GOES ON LINE

The Army Package Power Reactor, first complete pressurized water reactor plant for electric power, last month was taken up to full power, and then formally dedicated in the presence of AEC and Defense Department notables. Located on the bank of the lazy Potomac an hour's drive below Washington, in the Army Engineer Corps' Ft. Belvoir, Va., APPR-1 was begun Dec. 10, 1954, went critical April 8 and first generated electricity April 13. Five months were lopped off the schedule. Although the reactor was run at rated power—10 Mw heat, 2 Mw electric, much of the early testing was at 2 Mw heat, 500 kw electric. During these tests, NUCLEONICS learned, the reactor exhibited a degree of stability under violent load changes that surprised even its builders: when a circuit-breaker connecting the generator to the outside electrical system opened and load dropped instantaneously from 2,000 to 200 kw, the reactor followed within a minute and with no overpressure or over-temperature. The experiment was repeated deliberately, with the same result. Then load on the turbine was increased from 200 to 2,000 kw at a rate of 1,000 kw/min, and again the reactor followed smoothly. A 700-hr continuous-run test is now under way. The over-all cost of $3.5-million is being shared equally by AEC and the Army; of this $2,426,134 went to Alco Products Inc. for design and construction; other items include $330,000 for architectural engineering, $350,000 for the first six months' testing, and $120,000 for fabrication of the first core. Value of the fully-enriched uranium in it is placed at $382,000. Core life is 1½ years, in which time 7½ of 22½ kg U^{235} is consumed, for a 33% burnup rate. Operating cost figures were not disclosed: APPR's builders said that development of such information was one of the reasons for building the plant, and that no estimates had been or could be made until it had run for some time. Disassembled, APPR's fuselage-size components would make about 75–100 C-124 (Globemaster) loads, would take from six months to a year to assemble in the field. The plant output is 2,035 kw gross, 1,855 net, with 180 kw used for plant auxiliaries. There are 38 UO_2 fuel rods of the flat-plate type, 27 in. long, and seven control rod assemblies having a fuel section and a boron carbide poison section. Primary water under 1,200 psi enters the core at 431° F, leaves at 450° F; a U-tube type steam generator with built-in superheater makes 190 psia, 404° F steam. The turbine runs at 5,489 rpm; a 4.574:1 reduction gear steps down generator speed to 1,200 rpm. Alco built the reactor vessel, core basket, steam generator and feedwater heater; Oak Ridge National Laboratory the fuel and control rods; General Electric the turbine and steam generator; Westinghouse the primary coolant pumps; Lummus the condenser; Minneapolis-Honeywell the instrumentation; Bethlehem Steel the vapor container; General Ionics the demineralizers. Stone & Webster were engineer-constructors. The reactor will be used for training as well as to obtain operating experience; in January the first class of 30–36 officers and men from all three services will start six-month duty tours at the plant.

BLEEDER HEATER TESTS PWR RELIEF VALVES

Pictured is a high-pressure high-temperature bleeder heater, about to be shipped from the Massillon, Ohio, plant of Griscom-Russell where it was designed and fabricated. It was specially designed to be part of a high-pressure test loop to test PWR's relief valves, which are to operate at extremely high pressures. This heater is



capable of operating at pressures and temperatures on the shell side higher than any yet specified for feedwater heaters: 2300 psi and 940° F. The test loop will be set up at the Twin Branch plant of Indiana & Michigan Generating Co., where steam and water at required conditions can be supplied for the tests. This heater and another of conventional design will serve as part of a continuing testing program, to test other pressurized water reactor equipment for the naval program.

3 NEW FUEL FABRICATING FACTORIES

A $500,000, 32,000-ft² facility for making fuel elements has been started by General Electric at its Atomic Power Equipment Dept. in San Jose, Calif. An addition to the 15,000-ft² fuel development facility completed last year, this building is to be completed in December; it will contain $1-million worth of specialized machinery and tools, and will be used initially to develop and make fuel elements for Commonwealth Edison's Dresden Station. This is GE's first large-scale entry into the commercial fuel-element manufacturing field. . . . Two other firms are building plants for fuel element fabrication. Sylvania-Corning Nuclear Corp. (NU, Dec. '56, R6) will build a 25,000-ft² plant at Hicksville, L. I., adjacent to its present plant there, to produce fuel elements. It is to be completed by late summer. And D. E. Makepeace Co. of Attleboro, Mass., metal-working firm, is building a 53,000-ft² plant at Plainville near Attleboro primarily for working and treating uranium for fuel. It will produce fuel fillers, clad fuel elements, and related reactor components.

REACTOR FOR NEPAL?

A U. S. firm eight months ago offered to install a research reactor in Nepal, one of the world's most isolated countries. The government in Katmandu is "giving serious consideration to the offer," according to the Indian press.

World News

A-Ship Progress in Britain. . . .

Britain is planning a nuclear-powered tanker, to cost $30-million and take six years to complete at the Birkenhead yards of Cammell Laird. The Admiralty is expected to order a prototype. A Calder-type propulsion plant is considered unlikely, but Sir Christopher Hinton has suggested a D_2O-moderated, gas-cooled unit; Harwell has also been studying an organic-moderated marine power unit. . . . The Admiralty has awarded English Electric Co. a contract for design and manufacture of main propulsion turbines and condensers for Britain's first A-sub, HMS *Dreadnought* (NU, April '57, 21) and for its land-based prototype.

. . . and France

Natural uranium is to fuel the reactor in France's first A-sub, the *Q-244*, according to M. Baissas, assistant director of the Saclay nuclear center. The 5,000-ton craft, to be completed in four years, will serve as a training ship for crews of other French nuclear-powered vessels.

Germany Satisfied on Fuel Supply

West Germany's nuclear fuel supply is assured for several years at least, Atomics Minister Siegfried Balke declared on his return from a trip to the U. S., Canada and Britain. Agreements are being prepared covering delivery of 1,500 kg U^{235} from the U. S. and 500 tons natural U metal or concentrate from Canada. Balke also called for modification of German reactor plans to provide for supply of process heat.

New Industrial Groupings in Europe

Girding for the coming industrial drive in nucleonics, new combines are being formed at an accelerating pace. France's three biggest shipyards and the vast Schneider-Creusot steel-and-heavy-machinery group have set up a firm called Propatome to coordinate efforts in marine propulsion. The Banco Urquijo of Madrid and twelve associated Spanish industrial firms have established Tecnatom to develop nuclear energy in Spain in all its industrial applications. The German subsidiary of the big Swiss firm Brown, Boveri has set up a company called Atomkraft (Atomic Power) to plan, build and sell nuclear power plants. Another new German firm is the Society for the Development of Atomic Power in Bavaria, founded by the Bavarian state government, Farbwerke Höchst of Frankfurt, and four Bavarian electric utilities.

Australia a Factor in U Production

Australian production of uranium oxide will be 1,000 long tons/yr when the Mary Kathleen mine begins operations, according to Minister for National Development W. H. Spooner in the first Australian estimate ever disclosed. This rate of production is said to mean that Australia would overtake the Belgian Congo as a U-producer. Capital already invested in the U industry is officially estimated to total almost $70-million.

Heavy Water Made in Japan

Showa Denko Kaisha, one of Japan's top chemical fertilizer makers, has started operation of a D_2O pilot production plant, built at a cost of 117-million yen ($325,000), 81-million of which was met through a government subsidy. The factory is being enlarged; on completion in June it will produce 7–8 tons D_2O/yr.

More Details on RWE Plant

The first firm order for a West German nuclear power plant has been placed by Rheinisch-Westfälisches Elektrizitätswerk of Essen with the international combine of AMF Atomics of New York and Mitchell Engineering Ltd. of London, according to advices to NUCLEONICS from Germany (NU, March '57, R7). Total cost of the 15-Mw plant will be DM 30-million ($7-million), of which $5.3-million is for nuclear work. Of the latter figure, three-fifths will be earned in dollars by Mitchell for construction of the reactor vessel, heat exchangers and other nuclear equipment; AMF, responsible for system design, will supply the core, core control gear and instrumentation. Siemens-Schuckert of Erlangen is expected to provide the turbo-generator, alternators and switchgear. The reactor will be a closed-cycle boiling water type (*not* a pressurized water, as first reported), providing saturated steam at 900 psi, 530° F in the primary; 600 psi, 498° F in the secondary. Essentially it is a copy of the Elk River reactor design but without superheater, so it will have the same thermal output, 58 Mw, but only 15 instead of 22 Mw electrical. Moderator-coolant will be natural water. The plant is expected to start operation in the second half of 1959, producing electricity at 23 mills/kwh. Site has not yet been determined, may be underground. Seven other firms bid on the job, six of them U. S. companies, the seventh a German firm with American connections.

Offer to Denmark

The AMF Atomics—Mitchell combine has also made an offer to the Danish government, NUCLEONICS learns from Copenhagen, to build a 20-Mw nuclear power station in 36 months for $5.75-million. Price of power in the offer is reported to have been 10 mills.

India Seeks Critical Facility

The Government of India plans to set up a zero energy reactor at Trombay for studies in lattice spacing, fuel elements and reactor design. It is to cost $420,000, exclusive of heavy water (to be imported from the U. S.) and natural uranium; $1-million has been allocated for the entire project in the interim budget now before Parliament. Another $500,000 is budgeted for production of reactor-grade graphite, a zirconium plant and low-grade U ore beneficiation plants. The Atomic Energy Department intends to step up capacity of the rare earth plant in Kerala State from 1,500 to 3,000 tons monazite/yr, to nationalize the mineral sands separation industry there, and to set up another Th-U plant and a uranium metal processing plant at Trombay.

British Optimistic on Insurance

Third-party liability insurance on nuclear power stations will be readily available to "very substantial limits," according to an official of Alliance Insurance Co., London, who commented that the number of nuclear accidents are very few and that reactor operations have shown an accident ratio much lower than that expected in industry as a whole. However he said it was expected that fire insurers would not issue cover for damage to reactors by explosion or by self-generated heat.

Baghdad Training Center Opened

A joint nuclear training center set up with British help for the Baghdad Pact countries (Iraq, Turkey, Iran and Pakistan) was officially opened in Baghdad by Iraqi Prime Minister Nuri As-Said. It has a library and laboratories, will have a research reactor, and gives six-week courses to trainees from the four countries. It is headed by a British director.

News in Brief

New Faces at the Top

Col. James B. Lampert, head of Army Reactors Branch at AEC under whose supervision APPR-1 was successfully completed (*see p. 28*), will take up a new assignment at the National War College in late July or early August. His successor at AEC will be Col. Donald G. Williams, formerly on the staff of AEC's Military Liaison Committee . . . Alfred Iddles, who as president of Babcock & Wilcox since 1948 led the firm to a leading position in nuclear power, retired and was replaced by Morris Nielsen, executive vice president . . . Carroll V. Roseberry, first manager of commercial atomic power activities at Westinghouse, has been elected vice president and made manager of the firm's midwest region. New manager of CAPD—it has been renamed a department—is Robert L. Wells from Westinghouse's aviation gas turbine division at Kansas City.

Texas Utilities Organize Study

Eleven light and power companies in Texas have organized the Texas Atomic Energy Research Foundation. Its program will be directed toward the fundamental methods involved rather than to early use of nuclear power, said its head; "the program of the foundation is long-range, since the natural gas now used in the Southwest will be competitive with nuclear energy for many years."

Irradiated Coal: Diesel Oil Additive

The Denver & Rio Grande Western R.R., one of the few groups working seriously on atomic locomotive research, has been successful with another application of nuclear energy. It is using irradiated coal as an energy booster in Diesel engines. Low-cost Colorado coal is crushed to 200-mesh, sent to Brookhaven for exposure to spent fuel elements; under the gamma radiation the already-fine grains of coal are split into micron-size particles, making a coal dust that floats in air like smoke. Tests show that 5% of such coal added to each gallon of diesel oil increases its energy content. So far the tests were on a small 5-hp engine; full-scale tests on locomotives are now being scheduled.

Tracerlab Opens Big Plant

Consolidating eastern operations spread in seven plants, Tracerlab dedicated its $1.9-million building at Waltham, Mass., outside Boston. Housed in the two-story, 164,000-ft² plant are 15 major radiochemical and physics labs, manufacturing facilities for nuclear and X-ray equipment, engineering and research departments and sales and administration offices. Tracerlab also has a modern plant at Richmond, Calif., and a lab in Houston.

Industrial Briefs

Hughes Aircraft Co. of Culver City, Calif., has set up a new department of nuclear electronics, to study effects of radiation on electronic equipment. . . . NRC Metals Corp., the National Research Corp. subsidiary in which Columbia Southern Chemical Corp. bought a half interest (NU, April '57, 28), has been renamed Columbia-National Corp. . . . Horizons Inc. of Cleveland has gotten a contract from Argonne National Laboratory to develop a vessel impervious to molten fused salts, in connection with ANL's fused-salt work in developing a fluoride volatility process for reprocessing spent fuel. . . . AEC has given a one-year contract to Ramapo Uranium Corp. of New York, to buy at $8/lb U_3O_8 in uranium concentrates produced in the firm's pilot plant at Warwick, N. Y., up to a maximum of 50,000 lb. The plant will process U-bearing ores of the Warwick area. . . . Eitel-McCullough, San Bruno, Calif., maker of transmitting tubes, has entered the nucleonics field with receipt of an $860,000 AEC contract for research and development on an undisclosed project. Eimac expects the two-year job to lead to a production contract. . . . Wah Chang Corp., zirconium producers, are perfecting a new process to separate tantalum from niobium by a procedure simpler than the present liquid-liquid method. . . Westinghouse has installed two new test loops to further its attack on the corrosion problem. They can operate at up to 2,000 psi and 600° F, can each accommodate up to 210 specimens at a time. . . . W. R. Grace & Co. is starting work on a raw materials production plant at Erwin, Tenn., for uranium, thorium, and rare earth metals and alloys, says it will be the first such plant entirely privately-financed and -operated. Solvent extraction, reduction, melting and casting will be carried out.

Americans Participating in Japanese Meeting

Thirty or more Americans from industry, government, education and research will be speakers at the conference on peaceful uses of atomic energy sponsored jointly by the Atomic Industrial Forums of the U. S. and Japan. Sessions will be held in Tokyo May 13–16, and separate meetings May 17 in Osaka and Nagoya. Twelve or more of the biggest U. S. manufacturers of reactors and equipment will exhibit at shows in Tokyo May 12–19 and Osaka May 24–31. Representatives of other Far Eastern countries are expected.

Dates Set for 1958 Nuclear Congress

The third Nuclear Science and Engineering Congress will be in Chicago March 16–21, 1958. The National Industrial Conference Board will again synchronize with its Atomic Energy in Industry meeting, set for March 17–19 in Chicago.

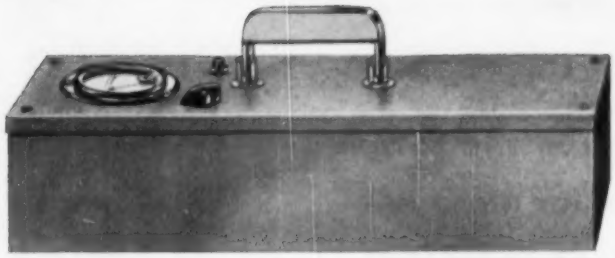
Babcock & Wilcox Reports Loss Year

Atomic energy business in 1956 cost Babcock & Wilcox a $2.4-million loss, its annual meeting was told. Retiring president Alfred Iddles declared, however, that the loss was justified by expectation of future profits. But it is impossible to predict when the profits will begin, he said.

Southern Regional Unit Organizes

The South's new 23-man Regional Advisory Council on Nuclear Energy, set up by the Southern Governors' Conference (NU, Sept. '56, R8), has named G. O. Robinson, assistant to the manager of AEC's Savannah River Operations Office, as executive secretary. Robert M. Cooper of Columbia, S. C., director of State Development Board there, is first chairman, and Dallas attorney Frank Norton vice-chairman. In a precedent-setting move, the 16-state body decided to contact all utilities, both private and public, in the area as well as others known to be interested in nuclear power development, to offer help in removing any restrictions on the state level that might impede its development.

New



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SPECIFICATIONS:

Slow Neutron Detector: Enriched BF_3 detector counts neutrons below 0.1 MEV.

Fast Neutron Detector: Methane-Argon filled, polyethelene lined detector counts neutrons with energies between 0.1 and 15 MEV.

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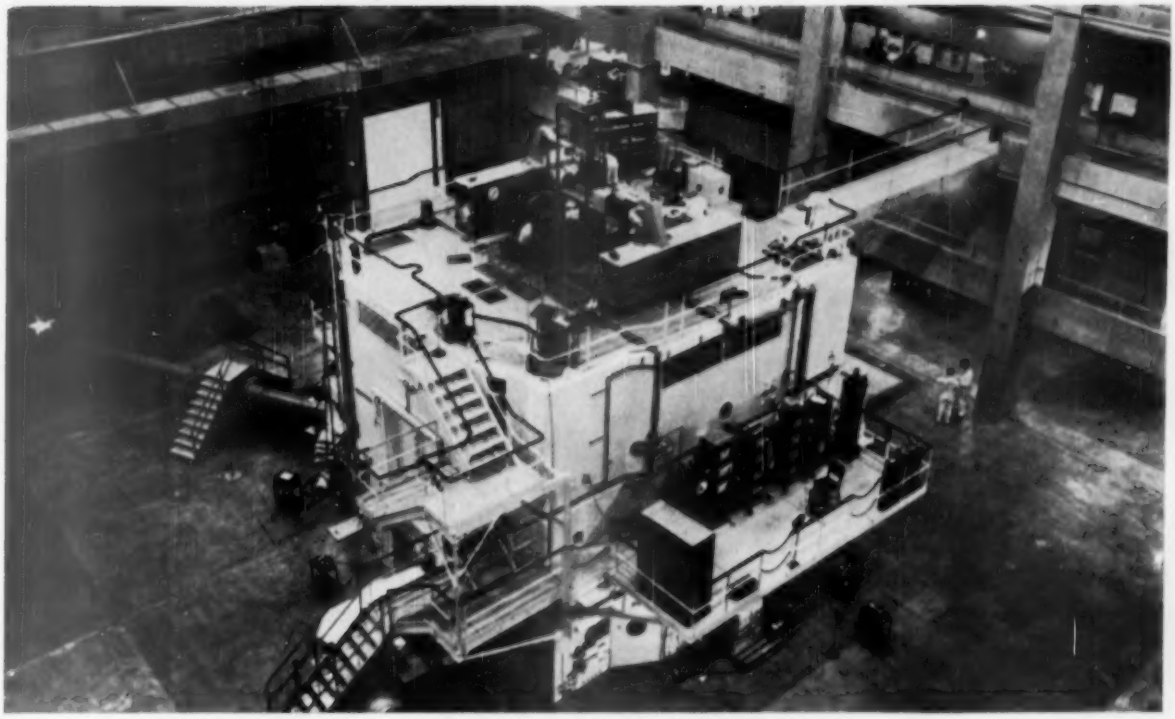

World's FIRST testing reactor—the Atomic Energy Commission's materials testing reactor at Idaho Falls, Idaho. Blaw-Knox served as architect-engineer. Neutron "chopper" tube is on main floor, left; control room is at end of bridge, upper right.

Blaw-Knox engineers two firsts in Atomic Testing Reactors

The need for testing reactors is so great that the Atomic Energy Commission has announced a new testing reactor policy, which encourages the building of testing reactors by private industry. As AEC states: ". . . a large part of its (the Commission's) requirements for test reactor facilities can be met with privately-owned reactors based on proved designs and technology."

Blaw-Knox has played a prominent role in developing this technology. Our experience includes serving as architect and engineers on:
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- summarizing hazards reports
- detailing engineering
- procurement and fabrication of parts
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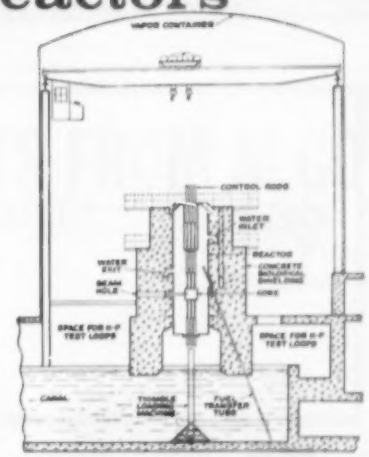
- initial operation
- training of operators

We welcome the opportunity to work with you on any proposed project in this active field of nucleonics.



Sectional view of privately-owned testing reactor, engineered by Blaw-Knox, shows general assembly, control rods and pressure vessel.

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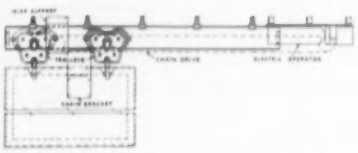
If you require lead bricks for constructing portable barriers or high density blocks for shielding enclosures, Ameray can supply you with the proper sizes, shapes and densities in materials to your specifications.



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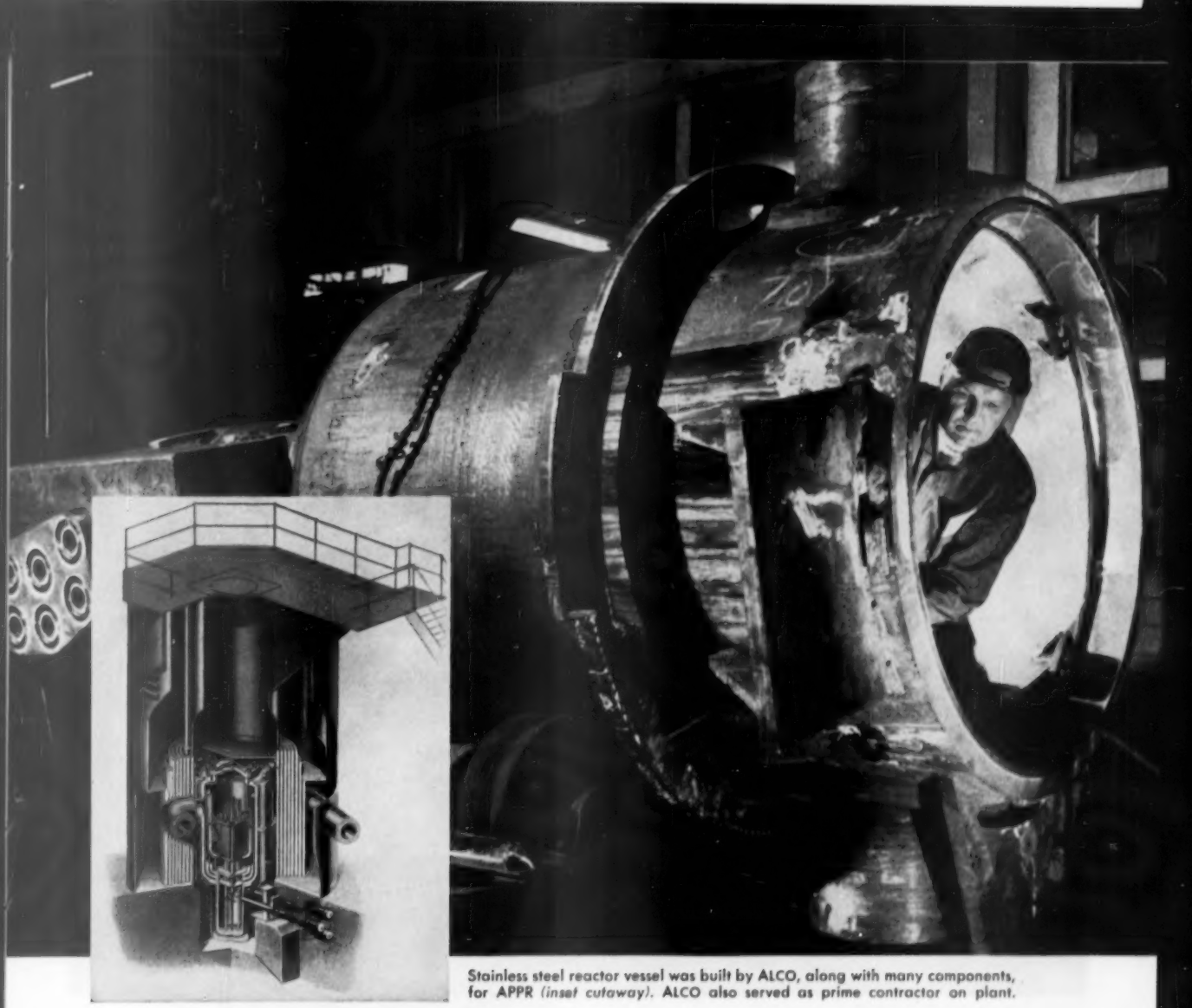

Stainless steel reactor vessel was built by ALCO, along with many components, for APPR *(inset cutaway)*. ALCO also served as prime contractor on plant.

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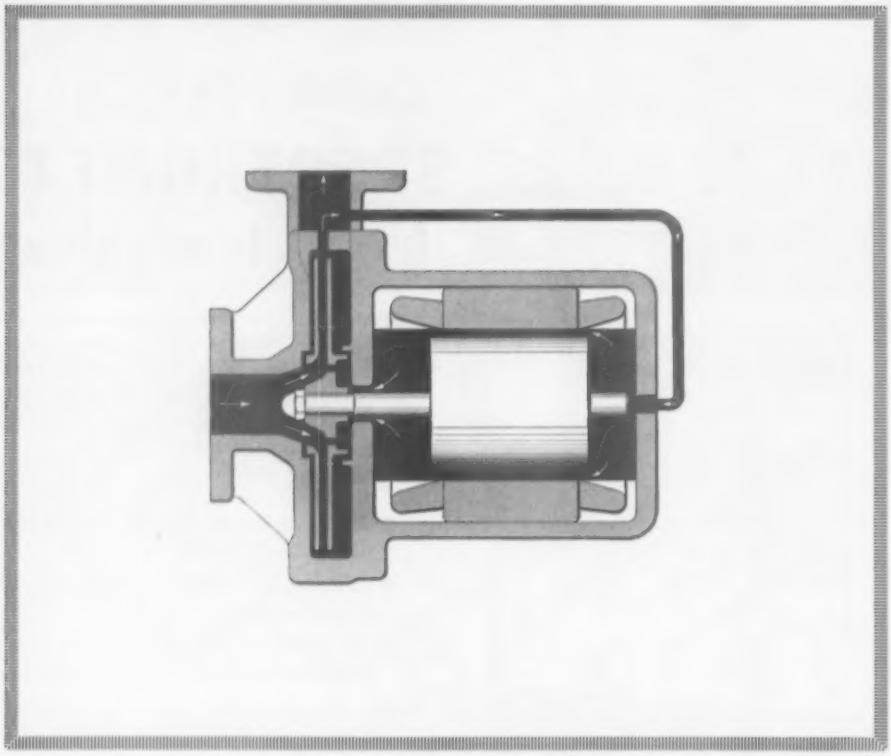
Typical scans made with Model 1820 Recording Spectrometer using DS5-3P scintillation detector with 2″ x 2″ solid crystal. Window width was ¾ volt during all four scans. Energy figures are in mev.



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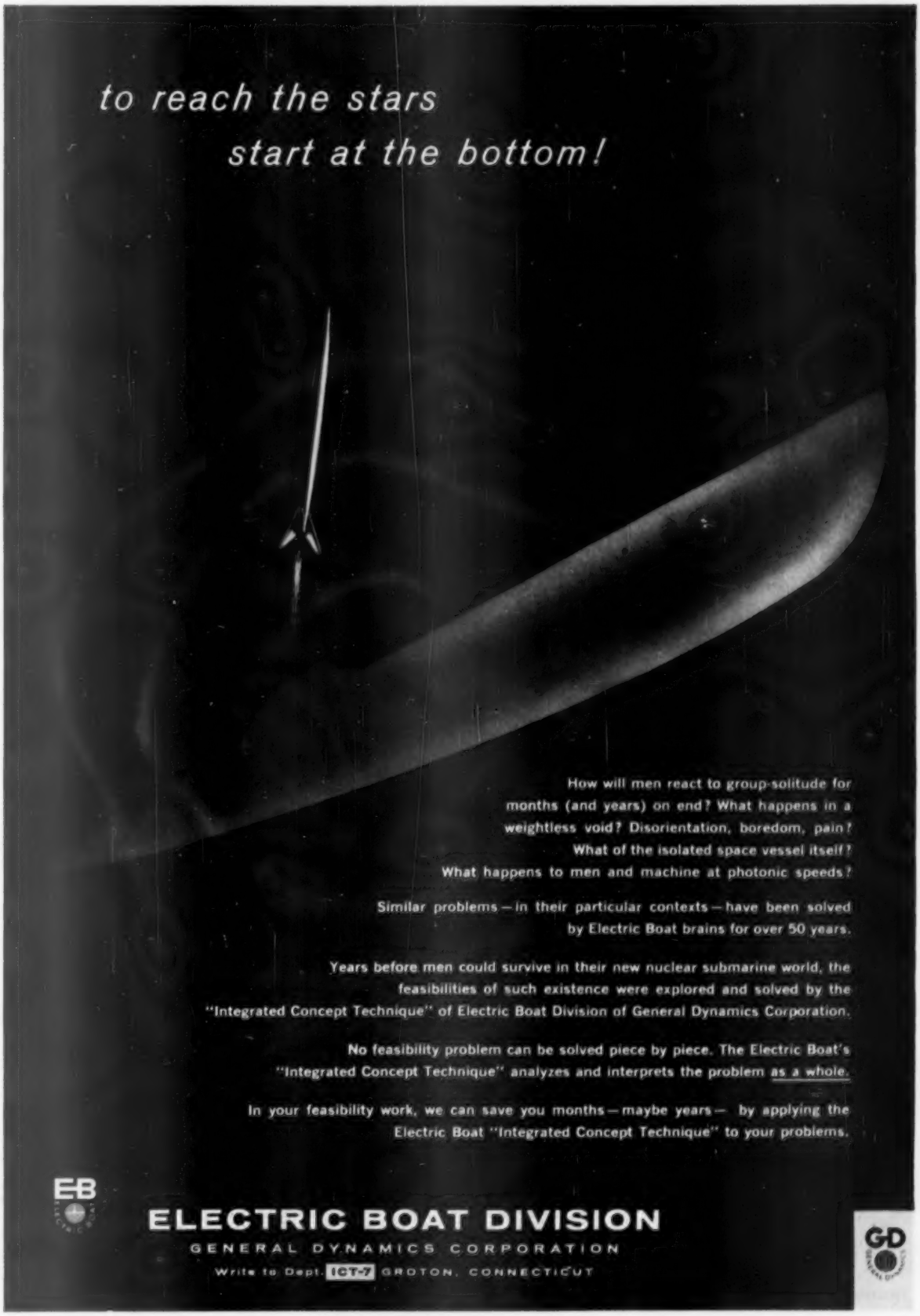

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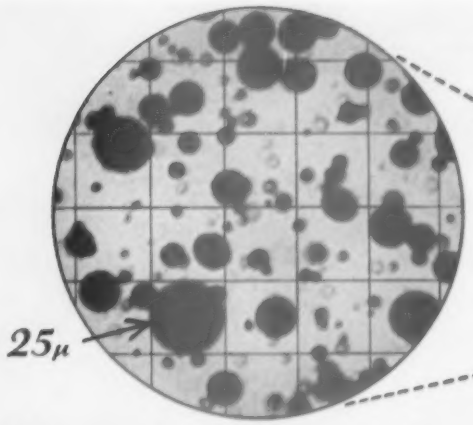

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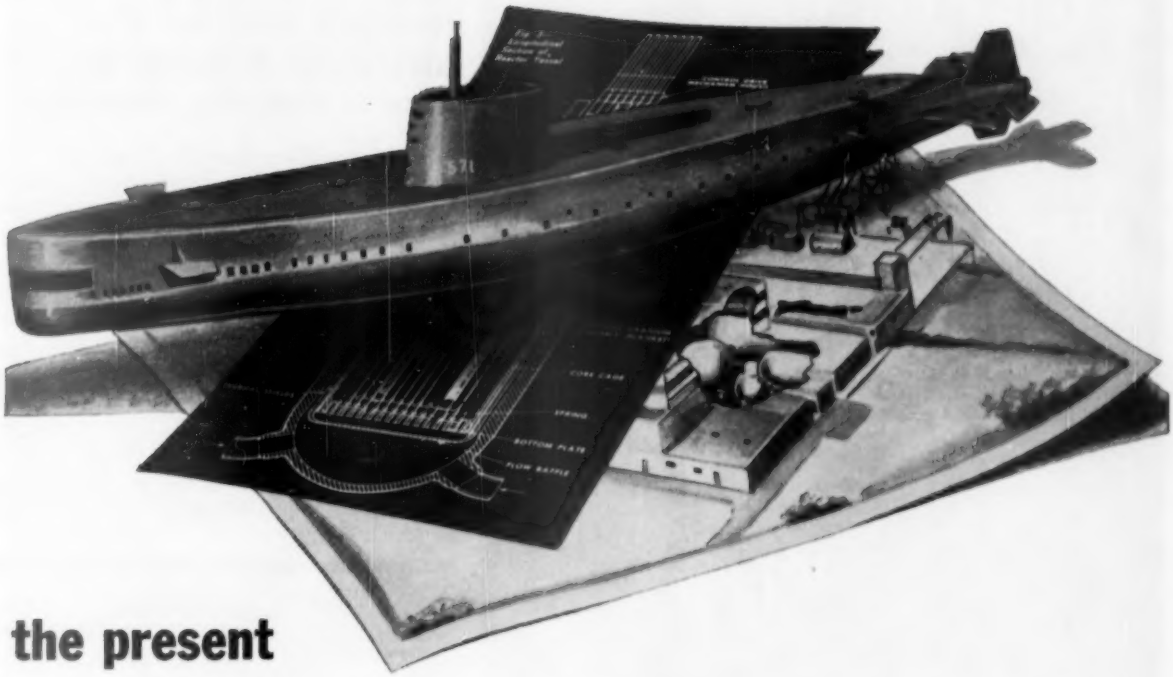


the present

and

YOUR FUTURE IN ATOMIC POWER

The USS *Nautilus* prototype was the first successful application of nuclear power. In 1957 the nation's first full-scale commercial generating plant at Shippingport will have its turbines powered by a Westinghouse reactor. Although the success of the nuclear reactor is a historical milestone, the application of nuclear power is still in the pioneering stages.

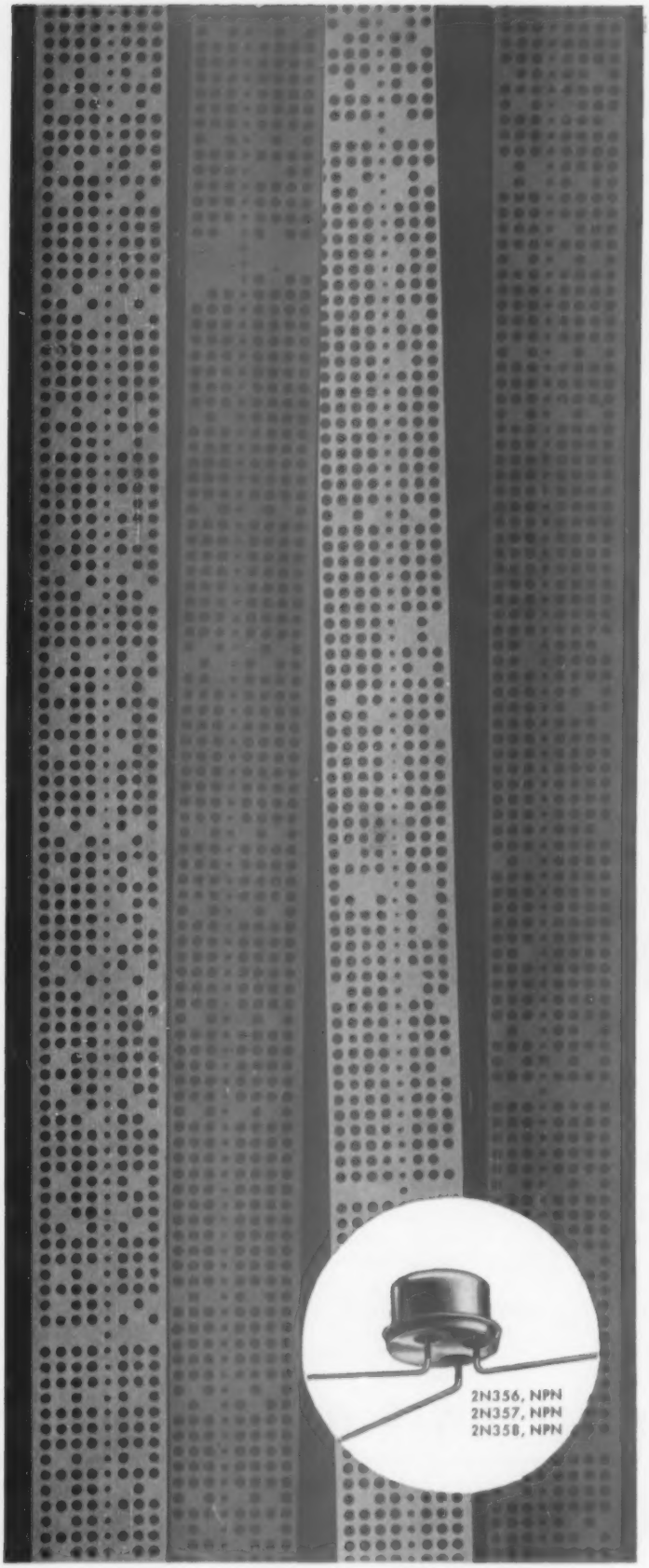
At Bettis Plant, operated by Westinghouse for the Atomic Energy Commission, more nuclear power reactors are being designed and developed. Here scientists and engineers are continuing to investigate new areas for progress in all phases of reactor theory, design, and application.

Much applied research remains to be done before the potentialities of nuclear energy can be utilized to the fullest. Opportunities for original work in a variety of fields present a creative environment for professional growth. Outstanding young scientists and engineers can also earn a PhD through a work-study plan in our doctoral fellowship program.

If you are a physicist, mathematician, metallurgist, or a mechanical, chemical, or electrical engineer, interested in the challenging opportunities offered by the leader in the nuclear power industry, write for a descriptive brochure on the opportunities in your field. Be sure to specify your interests. Address: Mr. M. J. Downey, Westinghouse Bettis Plant, Dept. A-154, P. O. Box 1468, Pittsburgh 30, Pennsylvania.

BETTIS ATOMIC POWER DIVISION

Westinghouse



General NEW HI-SPEED NPN SWITCHING TRANSISTORS Assures Computer Reliability

Computer engineers long seeking NPN transistors in applications requiring high current and fast switching will specify General Transistor's new 2N356, 2N357, and 2N358 for peak reliability.

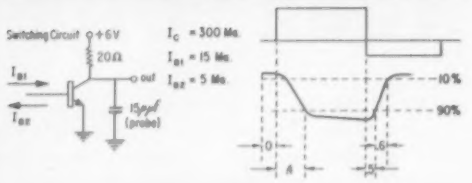
2N358: As developed by General, a typical switching speed of .4 of a microsecond at 300 milliamperes of collector current is possible with only 15 ma. of drive current.

The series resistance of these GT transistors, when conducting, is ½ ohm; the nonconducting series resistance is as high as 5 megohms with a result that approaches optimum efficiency at high current levels.

Computer manufacturers know they can depend on General's engineering and development as well as their quality and service. That's why GT is the largest supplier of transistors for computers.

2N358 CHARACTERISTICS

Parameter	Conditions	Min	Typical	Max
Collector-Base Voltage	Ic = 25μa	20V	30V	
	Vcb = 5V			5μa
Collector Cutoff Current	Ic = 300ma			
	Vce = .25V	20	30	50
D.C. Current Gain	Ic = 1ma		9mc	
Alpha Cutoff Frequency	Vcb = 5V			
	Ic = 1ma		9mc	



Write for GT's special NPN Computer Transistor Specifications Bulletin.

GENERAL TRANSISTOR CORP.

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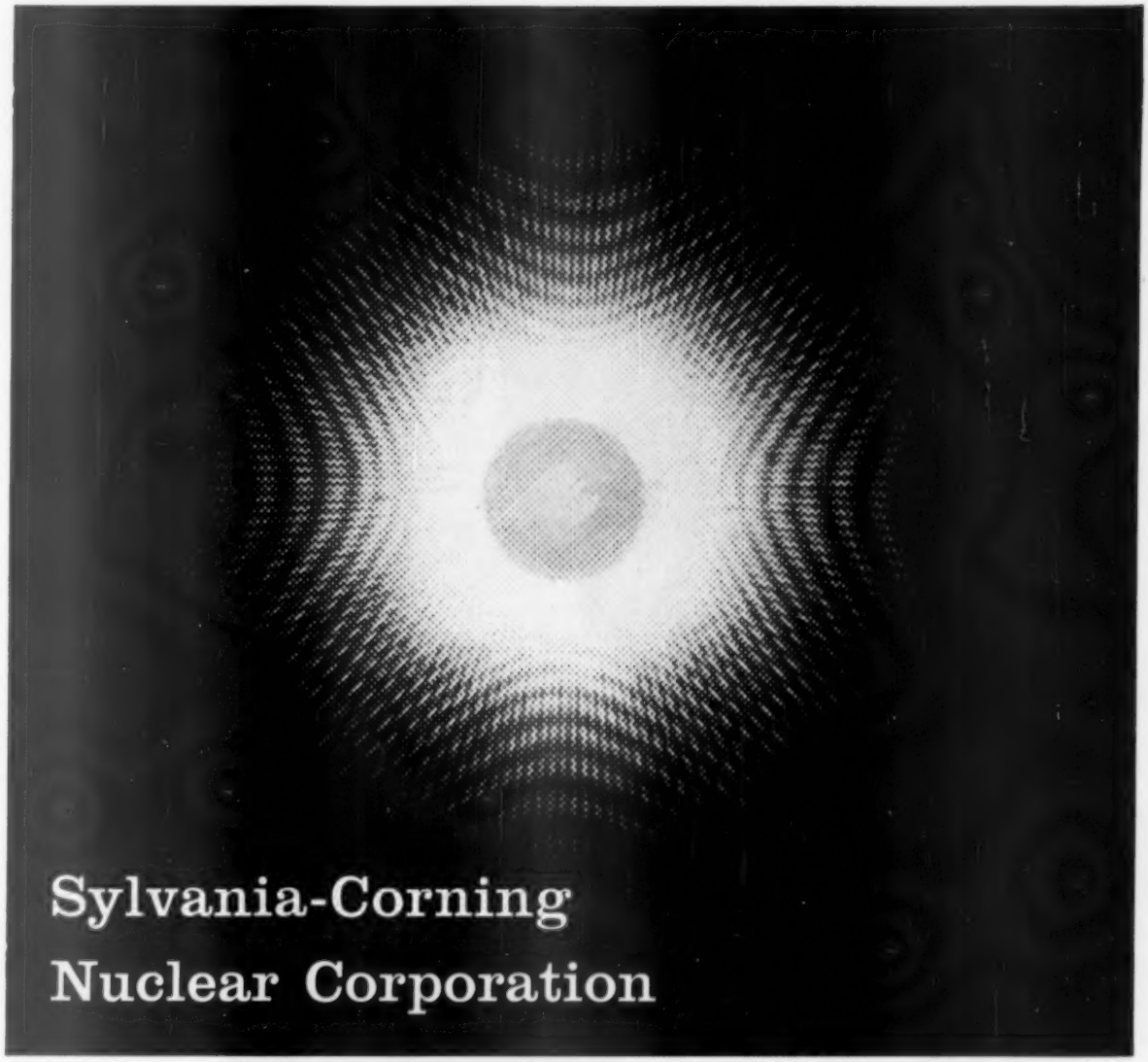


The world's first atomic fuel company

Now that the energy of the atom is being successfully harnessed to commercial uses, the world is rapidly moving ahead to exploit this huge source of power. Several nuclear power plants are already producing electricity. Many more are being built the world over.

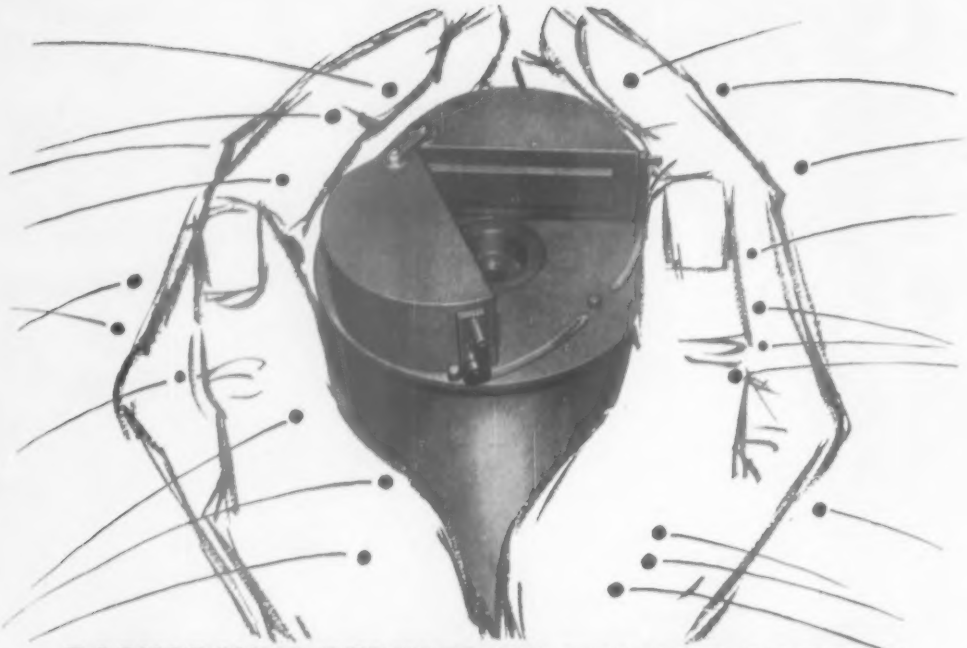
To supply the growing need of these atomic power plants for any type of nuclear fuel, Sylvania Electric Products Inc. and Corning Glass Works have formed the jointly owned Sylvania-Corning Nuclear Corporation. Through specialization in the design, fabrication and processing of nuclear fuels, the Corporation's aim is to provide complete and continuous service to all of the users of nuclear fuels.

Sylvania's Atomic Energy Division brings to the new corporation more than eight years of experience in the design and fabrication of atomic fuels and fuel recovery processes. Corning Glass Works has pioneered in the development of high-temperature ceramic nuclear fuel elements.

The nuclear energy activities of both companies have become the Sylvania-Corning Nuclear Corporation, the only company devoted exclusively to the development and supply of nuclear fuels.

For more complete information on this new organization, write to the address below. The combined scientific, engineering, and technical staff will be glad to help you solve any problems you may have in nuclear fuel application—whether immediate or in the planning stage.

SYLVANIA-CORNING NUCLEAR CORPORATION, Bayside, N. Y.



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. . . with new Tracerlab low background well scintillation counter - SC-57

Background of ~ 100 CPM is normal for SC-57, with its 1¾ inch diameter by 2 inch Well Crystal. This contrasts with the normal background of approximately 500 CPM with the same crystal in the SC-46. Users report a plateau slope of less than 1 percent per volt per 100 volts for a length of 200 volts, when used as a standard detector.

Versatile in application! The SC-57 may be used either as a spectrometry detector or as a standard detector for quarter-volt scaler imputs. An amplifier by-pass switch makes this dual use possible.

Long-term count stability is built into the SC-57 by mounting the phototube amplifier assembly on the outside of the well shield. This design feature provides absolute thermal and optical isolation of the photomultiplier tube from the amplifier vacuum tubes.

Low activity liquid sample counting is a particular area in which the SC-57 has already proved valuable.

Complete information on the Tracerlab SC-57 Low Background Well Scintillation Counter will be sent to you by return mail. Why not write today?

SC-57 Counter and "1000" Scaler provide a Tracerlab combination of instruments that is unparalleled for low background.

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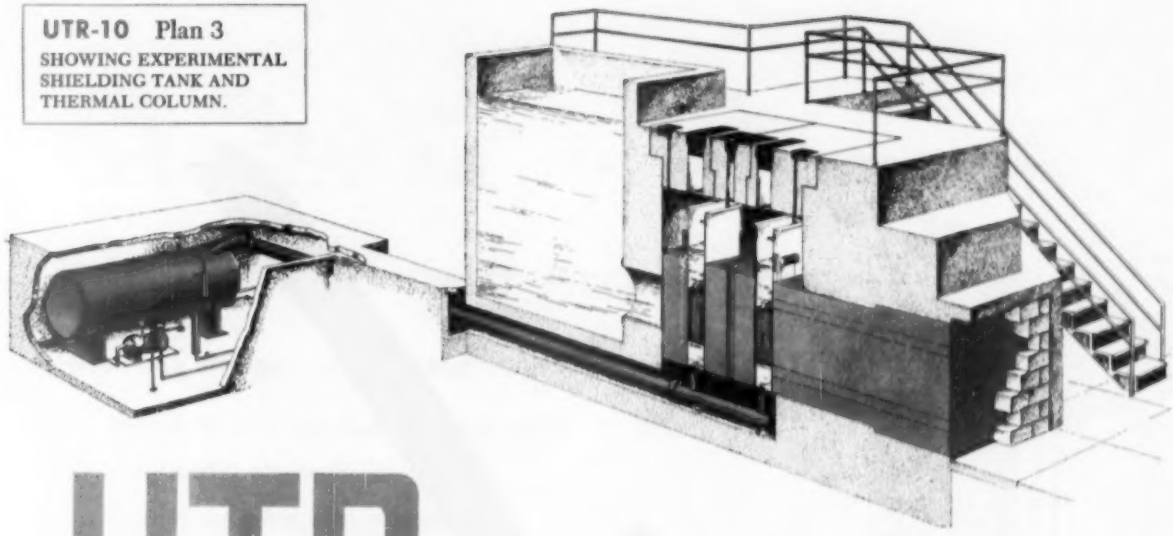

UTR

most neutron flux per dollar

American-Standard's University Training Reactor, a graphite-reflected, water-moderated tank type reactor, was developed from the proved training reactor concept originated in the prototype ARGONAUT at Argonne National Laboratory.

The UTR-10 design power level of 10 kilowatts was established on the basis of providing, at minimum cost, an optimum neutron flux level of 10^{11}n/cm^2-sec... required to meet the needs of a comprehensive nuclear training curriculum.

For details on the UTR program, including information on various plans to meet your specific requirements, write for the UTR descriptive brochure.

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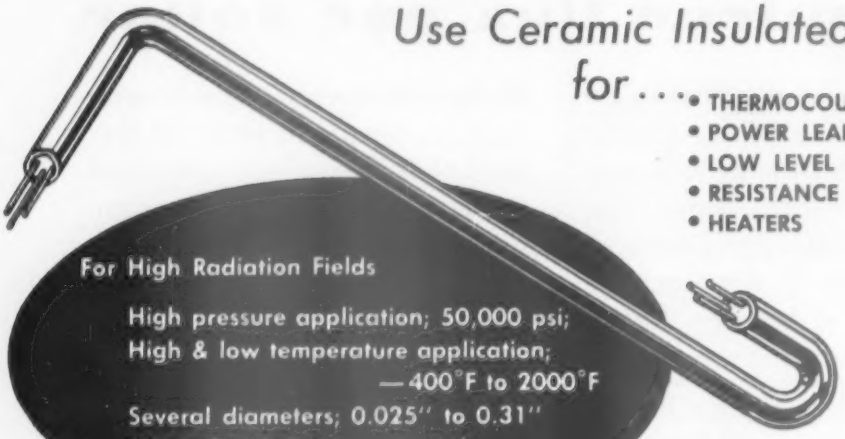

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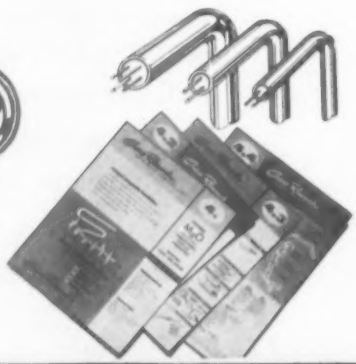

* Insulated wire consists of conductors, surrounded by ceramic insulation, with metal tubing overall.

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Briefly, these are facilities Arco offers:

Electron Beam Energy	2 to 10 Mev
Electron Beam Power	Variable to 4 Kw
Dose Rate	Variable to 1.6×10^6 r/min.
Average Beam Current	700 μa
Peak Beam Current	200 ma
Beam Pulse Length	10 μsec
Pulse Repetition Rates	1 to 360/sec
Scan Width	3 to 18 inches
Maximum Electron Penetration (±20% dose variation)	
One side	5 cm water
Cross-firing	8 cm water
Maximum X-ray Penetration from one side (±20% dose variation)	20 cm
Maximum X-ray Intensity (at 18")	1.6×10^6 r/hr
Maximum X-ray Beam Power (6 Mev electrons)	700 watts
Maximum Total Neutron Production	1×10^{11}/sec
Maximum Thermal Neutron Production	5×10^5/cm²/sec

Chemical laboratory, vacuum line, dosimeters, industrial gases, machine shop.

New brochure describing Service Center facilities sent on request.

New Arco Irradiation Center is at Walnut Creek, California, 25 miles from San Francisco.

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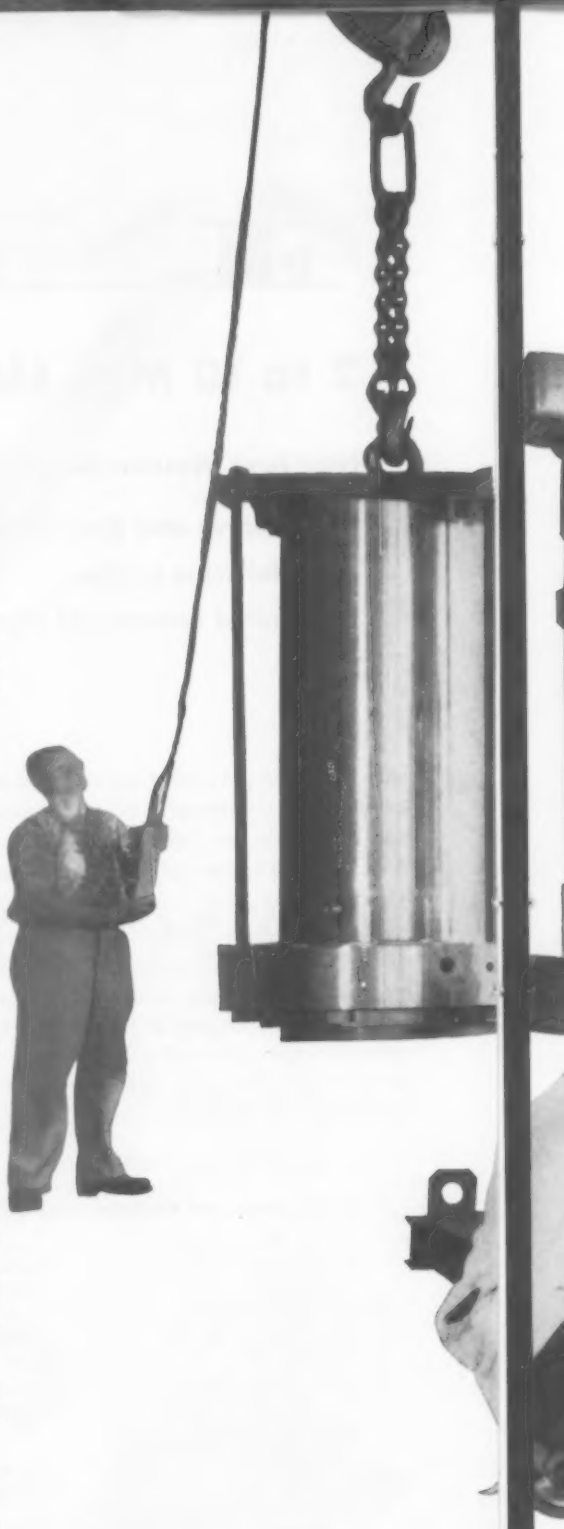
Atomic Fuels . . . complete cores for nuclear reactors are being manufactured in a new plant at Cheswick, Pa. The new facility has special machinery incorporating the latest design features for efficient high-quality production. Personnel qualified in the design and manufacture of cores for pressurized water reactors for power use are now available for advanced engineering assistance and consultation. Inquiries are invited for engineering and manufacture of all types of power reactor and test reactor cores.

Hermetically-Sealed Valves . . . basic designs for high-pressure, zero leakage applications are ready for development to customer specification in a range to fit ½-in. to 18-in. lines. Engineering assistance and application data for both pressurized water and liquid metal installations are available.

Westinghouse Leadership . . . gives you an unparalleled source of fully qualified products and advanced engineering assistance. Call your Westinghouse sales engineer or write: Westinghouse Electric Corporation, Atomic Equipment Department, Cheswick, Pennsylvania. J-57009-A



YOU CAN BE SURE...IF IT'S

These four canned motor pumps will handle 75,000 gpm of pressurized water in a large power reactor, now nearing completion.

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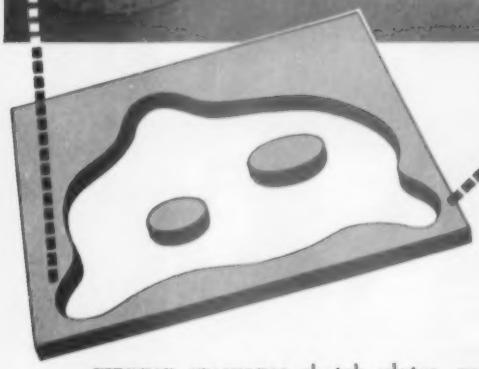

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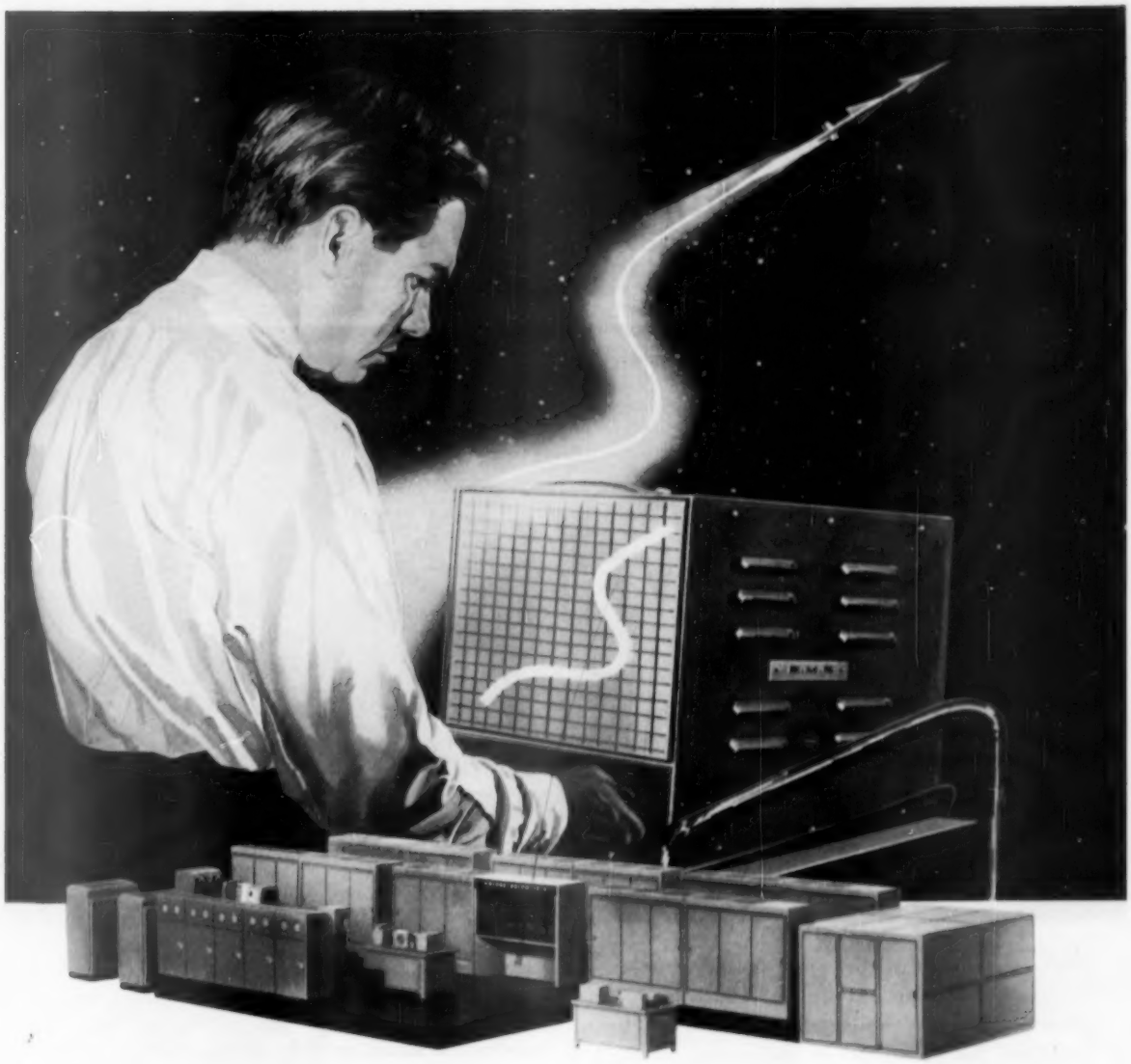

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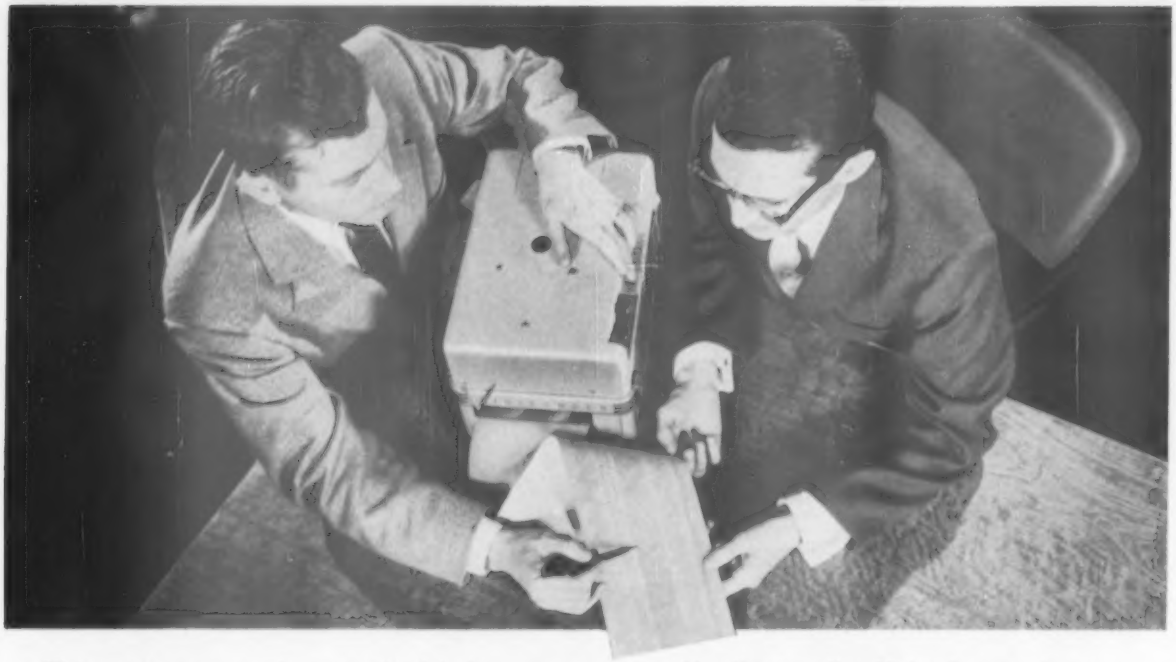

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also ANNOUNCING . . .

The high-frequency Visicorder galvanometers have been redesigned to provide sensitivity improvements as great as 4 times, and a new 1000-cycle galvanometer has been added. All high-frequency galvanometers shipped after March 15 are to the new specifications.

*Visicorder demonstrators are now based in these Honeywell Industrial Sales Offices: Albuquerque • Atlanta • Baltimore • Boston • Buffalo • Cleveland • Dallas • Dayton Denver • Detroit • El Paso • Hammond, Ind. • Hartford • Long Island City • Los Angeles • Omaha • Pittsburgh • Philadelphia • Richmond • San Diego • San Francisco Seattle • St. Louis • Syracuse • Toronto, Ont. • Union, N. J. • Washington D. C. • Amsterdam, Netherlands • and more on the way.

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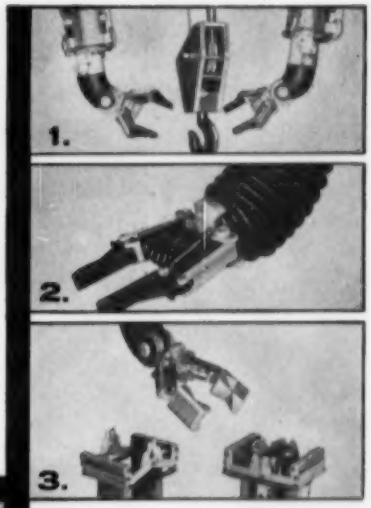
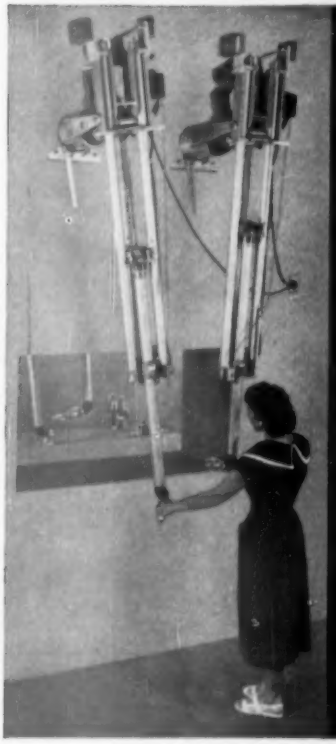
The "sense of feel", the versatility, the uncompromising workmanship . . . all are found in every CRL Manipulator. Research and production organizations all over the world are using these units for handling radioactive materials, explosives, and hazardous chemical and biological materials. Natural motions of the operator's hands are copied exactly at the slave end. The "sense of feel" helps the new operator to become perfectly competent with practically no training period. Handles are adjustable to comfortably fit a wide range of hand sizes.

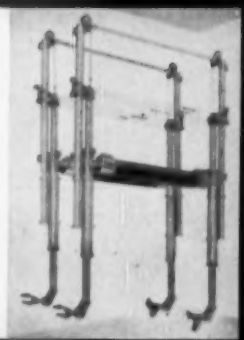




CRL Model 4

for over-wall installation

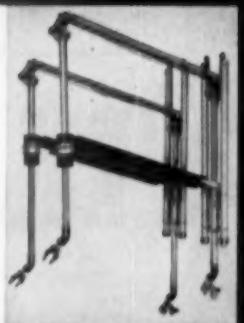
A general purpose manipulator for installation over a barrier wall or through a ceiling opening of a protective enclosure.



CRL Model 7

for restricted space installation

A space conserving, light weight manipulator for use in hoods or over thin walls where minimum clearances and easy portability are important factors.



1. *Cranes and hoists* For moving heavy loads in hot cell areas by remote control. They are specially designed by CRL for use with CRL manipulators.

2. *Flexible booting* Protects slave end from dust and contamination. Restricted gas-flow seal possible. Won't interfere with manipulator's natural movements.

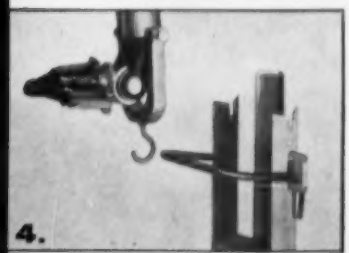
3. *Special-purpose tongs* For repetitive or highly specialized operations. Wide selection of types to meet all requirements.

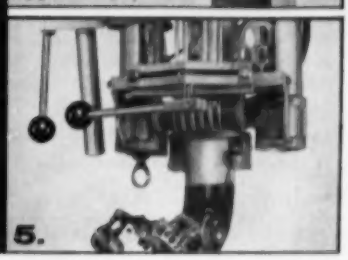
accessories

for master-slave manipulators

4. *Load hook* Available for CRL Model 8; may be remotely attached or removed. Permits a vertical lift of 60 lbs.

5. *Motion locks* Permit objects to be rigidly positioned in space. *Locking grip* standard on all models. Locks for all other motions available as accessories.





CRL Model 8

for thru-wall installation

A general purpose manipulator for installations requiring intermediate to high level shielding, the Model 8 features

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- *Greater versatility* A variety of tools, tongs and locking motions available
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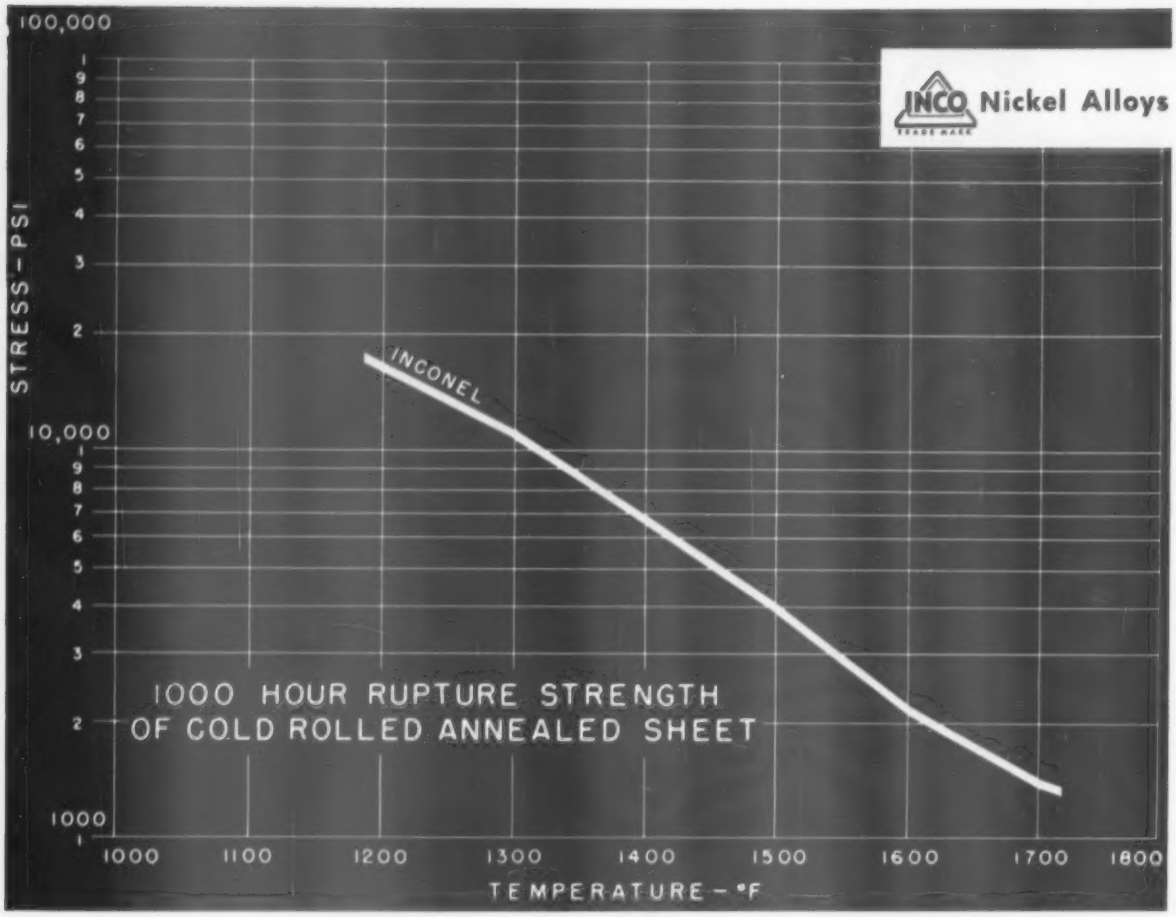

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Need metal that resists deformation under continued high stress at high temperatures and in corrosive atmospheres?

You have it in Inconel* nickel-chromium alloy.

Inconel alloy is unique in its high performance characteristics at elevated temperatures. It is widely used in heat-treating, in high-temperature chemical processing, in jet aircraft engines.

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Inconel alloy is structurally stable. It is not subject to embrittlement on prolonged heating in non-sulfidizing atmospheres.

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Inconel alloy is mechanically stable. It retains useful properties to 2100°F., and over, in some applications, and withstands thermal shock.

Inconel alloy is easy to work . . . easy to form, machine, weld . . . available in all commercial wrought forms.

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For complete information on Inconel (and age-hardenable Inconel "X") alloy, write for Inco's 24-page Technical Bulletin T-7.

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Non-magnetic

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Specific Gravity	8.51
Melting Point °F	2540-2600
Specific Heat (32-212°F) Btu/lb/°F	0.11
Thermal Expansion Coefficient (32-212°F) 10^6 in/in/°F	6.4
Thermal Conductivity (32-212°F) Btu/sq ft/hr/°F/in	104
Electrical Resistivity (32°F) ohms/cir mil ft	590
Tensile Modulus of Elasticity 10^6 psi	31
Torsional Modulus of Elasticity 10^6 psi	11

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Cold rolled strip	Extruded tubing
Hot finished rods & shapes	Hot rolled plate (solid)
Hot rolled angles	Forged billets & shapes
Cold drawn rods & shapes	Hot rolled sheet
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Condenser tube	Cast fittings
	Hot rolled plate (clad steel)
	Wire

Inconel . . . *for long life at high temperatures*

A NUCLEONICS SPECIAL REPORT

Analog and Digital computers in Nuclear Engineering

Analog and digital computers are widely used in nuclear engineering—from fundamental studies of cross sections to applied studies of pipe stresses, from figuring fission-product buildup over years to microsecond neutron histories. We have tried, in assembling this special report, to reflect that variety—the first five articles are on digital computation, the last six on analog. In both sections we have gone from an introductory primer to a general survey. The remaining articles tell of specific applications, chosen for their newness. This special report by no means exhausts our interest in computers; we intend to continue to report on developments in this field.

—The Editors

An Introduction to Digital Computers

Input, memory, control, arithmetic and output work with superhuman speed and accuracy to solve complex problems digitally

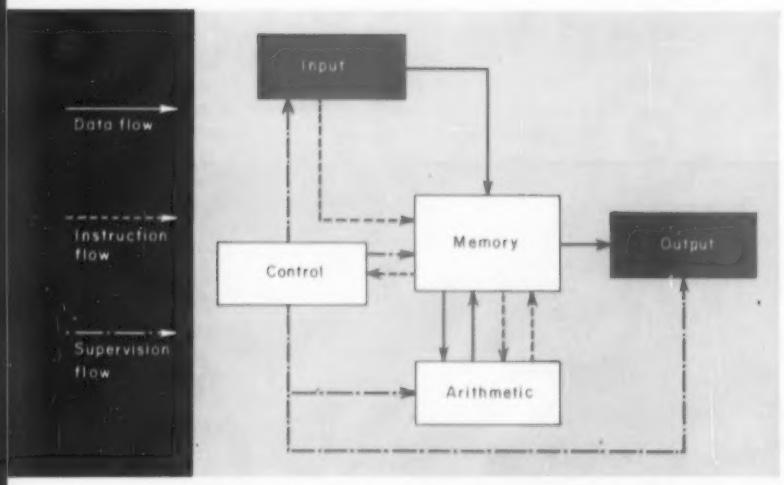

FIG. 1. Information flow among the five sections of a digital computer

DIGITAL COMPUTERS* are designed to contribute superhuman speed and accuracy to the solution of mathematical problems. They perform basic arithmetic operations and prescribed logical tests, all based on simple additions and subtractions. They cannot be the "brains" of an engineering department. Rather they are extremely efficient mathematical robots that execute instructions blindly and faithfully.

Spectacular developments in electronics during the 1940's provided the components and techniques that made feasible the development of high-speed electronic computers. This early period saw the outbreak of a rash of custom-built, one-of-a-kind computers like ENIAC, SEAC, SWAC, etc.

Simultaneously other engineers and scientists were making progress in punched-card techniques. These were people who did not have the resources required for truly large-scale electronic computation but who already had punched-card equipment as tools for their accounting groups. The logical outgrowth of the use of this machinery for computation was the card-programmed calculator. This connected several accounting machines to make a single unit and resulted in a device suitable for engineering and scientific problems. Over 200 installations of card-programmed calculators were placed in the field, and over 2,000 punched-card calculators are in use now.

Today we are in a truly significant era of both equipment development and application development. This is because there are available many production-model computers for the engineer. More than 175 large-scale computers and more than 700 "intermediate" machines are presentlyinstalled.

Solving a Problem

The solution of an engineering problem with a digital computer consists of:

1. Formulation—mathematical expression of the physical situation
2. Numerical analysis—development of an arithmetic means of solution
3. Planning for the machine—development of an over-all strategy for using the machine
4. Programming—preparation of detailed instructions for the machine
5. Debugging—discovering and correcting errors in steps 1–4

* This article is based on material provided by Computer Usage Co., 18 E. 41 St., N. Y., N. Y.

6. Running
7. Analyzing results

These steps are not independent. For example, when interpreting results one must be cognizant of the numerical approximations developed in step 2.

Logical Organization

Each digital machine has its distinctive characteristics. However, they all have basic similarities in general logic. Figure 1 indicates the typical relationships among the sections:

Memory. This stores numbers and instructions coded in numerical form.

Arithmetic section. Here are performed the arithmetic processes.

Input. This provides the means for inserting both data and instructions.

Output. This presents the results.

Control. The control section interprets instructions and causes the machine to perform specified operations using specified data. It controls what the machine does, what data it uses, and in what sequence it operates. This is accomplished through interpretation of sequences of instructions.

Instructions, which are coded in numerical form, can be modified arithmetically by the machine. That is, the program can include instructions that will cause the machine to reinstruct itself. For example, if the result of some process is unsatisfactory—too large, too small, etc.—the machine can be programmed to vary the parameter in a systematic way until a satisfactory result is obtained.

Memory

In all currently operating machines the memories depend ultimately on binary representation. This comes from the ease of construction of two-state devices in which one state represents the "0" of a binary digit system and the other state represents the "1."

Physically one can form a binary element in many different ways. Common examples are (a) a hole in a card or paper tape, (b) a magnetized spot on a magnetic tape, drum, or disk, (c) a magnetized core, (d) a spot on a cathode-ray-tube screen, (e) a pulse in an acoustical delay line, (f) the charge on a capacitor, (g) a conducting (or cutoff) vacuum tube.

When it is desired in the computing process, it is possible to form decimal representations from binary ones. Then the decimal number can be recorded and stored with devices designed for the purpose. For example, this is done by punching 1 of 10 possible positions on a card.

Access time. Such devices as magnetic cores and cathode-ray tubes provide electronic speeds in the retrieval of required information. Such a retrieval requires only a few microseconds. The magnetic drum, disk, and tape, on the other hand, require physical movement of the magnetic surface relative to the reading head before the information can be obtained. This usually results in access times of at least several milliseconds.

In Fig. 2 we have an example of access times—those of a magnetic drum.

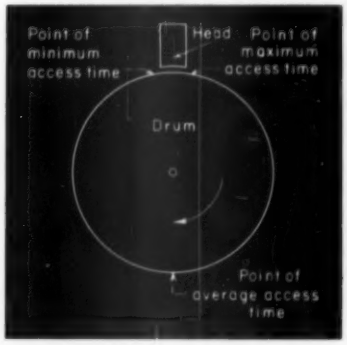

FIG. 2. Access time for a rotating drum depends on the storage position of the required word

If we want information that has just passed under the head, we must wait for a complete revolution of the drum.

Several techniques are used to improve the speed of such memories as the rotating drum. Optimal programming is an example. The programmer locates information with discretion so that it is passing under the head at opportune moments. Another means is a revolver, which reduces the storage area available and the maximum access time by the same factor. Head No. 1 continuously reads, and the information is amplified and rewritten on the track by the number 2 head. Information is stored only between the heads.

Arithmetic Section

All digital machines have arithmetic sections. Most have facilities for addition, subtraction, multiplication, and division, but some do not have division, and some have neither division nor multiplication. Most machines are symmetric, which means that if they accept word length a at the input and provide word length a at the output, the arithmetic unit has length $2a$. A "word" is a sequence of digits expressing either instructions or data.

Versatility is enhanced in many machines by providing for a floating decimal point. The first two decimal digits of a word tell the position of the point. They are written to equal 50 plus the appropriate power of ten so that they represent both negative and positive powers. Thus 12345.678 is expressed as 0.12345678×10^5 and written 5512345678, and 0.00012345678 is written 4712345678. The two digits that represent the exponent are called the "characteristic" of the number and the rest the "mantissa" or "fraction."

Input and Output

Of several types of input-output media in use today, the principal ones are punched cards, punched tape, and magnetic tape.

In these parts of the machine an important concept is buffering. The reading of the next card is anticipated so that its contents already reside in the buffer. When the instruction "read a card" occurs, the information on this card, having already been read into the buffer, is merely transferred from the buffer memory to the main memory. The computer goes on working, and the information on the next card is transferred to the buffer for the next "read-a-card" command.

Control

The control section of the computer interprets instructions, which are usually in numerical form. A typical stored-program instruction is made up of two parts. The two-digit operation code specifies the operation. Thus 01 might be addition, 02 subtraction, 18 multiplication, 29 division, etc. The address digits specify the operand to be used when performing the instruction. They give its location in the memory. A single-address instruction presumes a register in the arithmetic section holding the number on which the operation is to be performed. Thus a single-address instruction means "take the number in the register; operate on it with the number stored at the address given; and leave the result in the register." Other instructions must be used before and after this one to get the original number into the register and take the final result out.

The Role of Digital Computers in Nuclear Design

High-speed digital computers are essential to the modern nuclear reactor designer. The many programs coded for nuclear problems illuminate questions of application and cost

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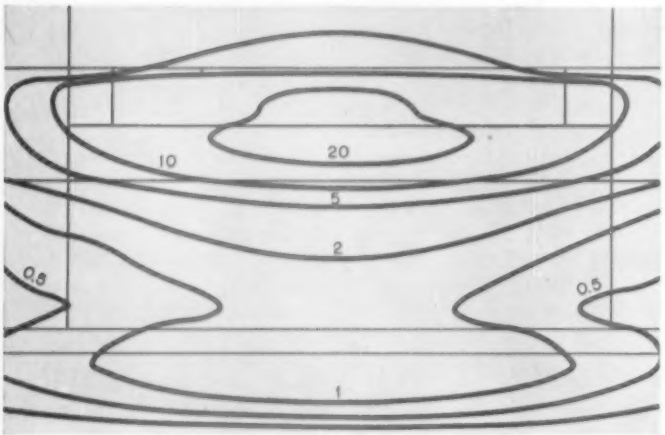

NEUTRON FLUX plot in two dimensions for Belgian Engineering Test Reactor typifies computer use in nuclear design. Central axis of reactor core is at top of figure and outer surface of water-shield annulus is at bottom. Top and bottom core-support plates are at right and left. Calculation by Nuclear Development Corporation of America using Datatron and new *R-Z* reactor code

THE USEFULNESS OF digital computers for nuclear calculations was appreciated early by scientists and engineers connected with atomic laboratories. Indeed, some individuals, such as John von Neumann, played major roles in the early development of both high-speed digital computers and nuclear energy. Calculating punch-card machines, such as IBM 602, 604, and Card-Programmed Calculator, were used for reactor calculations in the late 1940's. Oak Ridge National Laboratory, Los Alamos Scientific Laboratory, and Knolls Atomic Power Laboratory can be singled out for their early interest in such calculations.

There is little question today that nuclear reactor calculations on high-speed digital computers are vital, and will become more so, in furthering the design of nuclear reactors. Computing and mathematics groups are becoming a more essential part of the scientific or engineering installation. The recent establishment of these groups as major divisions at Brookhaven and Argonne National Laboratories emphasizes this trend.

Problems and Codes

Nuclear reactor calculations that have been programmed for digital computers are already so numerous and diverse that classification and cataloguing of the existing digital codes is not a trivial undertaking. In fact today this is equivalent to classifying nuclear reactors and all the problems involved in their design. Among the possible classification systems are those based upon the particular mathematical or numerical technique that is used; the number of independent variables, such as space dimensions; the aspect of the reactor that is being calculated; and the scientific or nuclear field from which the calculation arose.

Radkowsky and Brodsky (*1*) have listed and classified the digital machine codes used for reactor calculations that existed up to October, 1955. In this bibliography, reactor codes are classified under reactor physics, reactor survey, shielding, reactor kinetics, reactor burn-out, reactor engineering, or miscellaneous. The Newsletters of the Nuclear Codes Group (*2, 3*) have continued with the classification system of Radkowsky and Brodsky.

The classification below closely follows that of Radkowsky and Brodsky. A few comments have been added to

indicate the scope of each category.

Engineering. Calculations in this category involve engineering problems that do not involve nuclear considerations such as heat conduction, stress analysis, and flow problems. Although the most important constraints on reactor design are probably in engineering rather than in physics, relatively few codes have been written in this category. For example, even though heat limitations are of vital importance for any reactor used as a heat or power source, very little effort has been expended on codes for these problems.

Physics. Included is any calculation arising in physics, and particularly in nuclear physics that is related to reactors. In particular, most of the preliminary calculations of parameters that are needed in the other categories are placed here (e.g., the evaluation of resonance absorption integrals to provide effective multigroup cross sections).

Shielding. These calculations are concerned with the shielding of personnel and equipment from neutrons and gamma rays. Although adequate but not excessive shielding is needed for stationary reactors, most shielding codes seem to have arisen in connection with mobile reactors, where weight is of paramount importance. These shielding calculations fall roughly into one of three mathematical categories: Monte Carlo, numerical quadrature, and numerical solution of an approximation to the Boltzmann equation.

Short-term operability. This category includes kinetic calculations, temperature-coefficient calculations, and any similar "short-term" calculations pertaining to reactor operation. These codes help determine the safety, stability, and design parameters of reactors. Most of them investigate the kinetic response of reactors to step or ramp changes in reactivity. The reactor is usually considered to respond to changes as a unit, with time as the only independent variable. The resulting equations are ordinary nonlinear, differential equations. In solving these types of problems on analog computers, it has been helpful to observe immediately the changes brought about by changes in the input parameters. This fact suggests that digital computers with immediately observable outputs, such as the cathode-ray-tube outputs of the ORACLE and IBM 704, could be profitably used in a similar manner.

Long-term operability. This category involves calculations of fission-product buildup, burnup of fuel and poisons, breeding or conversion ratios as a function of exposure, and core-loading life. A series of criticality calculations with appropriate intermediate calculations can be used for "long-term" codes.

Diffusion and age-diffusion. This is the largest category of reactor codes and has as its starting point the diffusion and age-diffusion equations, with various alterations. The neutron fluxes and the critical mass or radius are what is desired. The age-diffusion equation can be derived from the transport equation by expanding the angular fluxes in spherical harmonics. The age-diffusion equation results from retaining only the first two terms in this expansion. Because of the similarity to retaining the first two terms in a Legendre series expansion, this is called a P_1 approximation. P_3, P_5, and higher approximations can be obtained similarly.

Transport approximations. These codes start with the Boltzmann transport equation and solve for the neutron fluxes and some critical quantity. The Boltzmann equation is a complicated integral-differential equation expressing the conservation of neutrons. The primary unknown function, the angular distribution of neutrons, can be a function of three space variables, three velocity variables, and one time variable. The physical or mathematical problem is completed when certain constants, primarily cross sections, and boundary conditions are prescribed.

Others. This category includes reactor chemistry, biology, mathematics, and codes that do not fit into the other categories.

Even with this classification system, it is very difficult to discuss comprehensively the nuclear codes that have been prepared. Among the contributing difficulties are (a) the security classification, in most cases unnecessary, of the codes and descriptions of the codes; (b) the lack of published and distributed write-ups of codes; (c) the inadequate interchange of information which has existed between workers preparing codes; (d) the newness and unsystematic development of the field; and (e) the almost continuous changes undergone by even the more successful codes.

However the bibliographies (*1–3*) of the nuclear codes, in addition to providing specific information about each of the available codes, do give a good feeling for the magnitude and distribution of the effort so far. The total number of codes listed in these sources comes to about 140. The breakdown by category is as follows:

Engineering	8
Physics	36
Shielding	10
Short-Term Operability	7
Long-Term Operability	8
Diffusion and Age-Diffusion	61
Transport Approximations	10

Even a casual glance at the distribution of codes shows that the greatest amounts of effort and time have been spent on solving the neutron-transport problem. More than half of all the codes are listed under the diffusion and age-diffusion category or under transport approximations. Within this family the various multigroup codes represent a particularly large investment of time and money. The two-dimensional code CURE, for example, involved over 8,000 man-hours and 70 machine-hours on the

Nuclear Code Fraternities

The Nuclear Codes Group, in existence since 1956, provides for interchange of information about nuclear codes on a nation-wide basis. At present the group has members from over forty installations, including nearly all companies and government installations interested in reactor design. Roughly half of these installations own, rent, or have ready access to large machines, and roughly half have medium machines available.

The group publishes a quarterly Newsletter containing information on the status of reactor codes at various installations. Specific information on a given program can be obtained directly from the originator or from the local representative of the computer manufacturer concerned. In addition general meetings are held semiannually concurrently with those of the American Nuclear Society. Inquiries should be directed to the chairman of the group, R. J. Graydon, Atomics International, Box 309, Canoga Park, Calif. Specific requests about the Newsletter should be sent to S. Schechter, New York Univ. AEC Computing Facility, 25 Waverly Place, New York 3, N. Y.

Two other organizations, Share and Use, made up respectively of users of the IBM's 704 and Remington Rand's 1103, have also given nuclear codes consideration. Share, the larger organization, contains about 70 installation members, of whom about 10 are concerned primarily with nuclear energy.

IBM 704 preparatory to running a production problem.

Neutron-Transport Codes

Because of the special emphasis the neutron-transport codes have received, a few comments on the actual techniques involved in these calculations are in order. The usual objective of any of these calculations is to obtain the neutron fluxes and the critical parameters for a given reactor system. The over-all mathematical technique for calculating a reactor can be only indicated here. The article by Ehrlich and Hurwitz (*4*) illustrates the multigroup method for solving the age-diffusion equation, and the report by Carlson (*5*) illustrates a method for solving the transport equation. In mathematical language, the problems are reduced, in the steady-state cases, to finding the fundamental eigenvalue (a critical parameter) and eigenfunction (the neutron fluxes) of homogeneous differential-integral equations. Numerically, it is necessary, of course, to reduce the problem to finding the fundamental eigenvalue and eigenfunction of associated matrices. Although the calculations and codes actually deal with matrices, it is not particularly advantageous to think of the problems in such terms. The power technique of matrix theory is customarily used to find the eigenvalue and eigenfunction. In the language of physics, this power technique is equivalent to first guessing a source due to fission, and then computing—in continuing sequence—fluxes from a source and a source from fluxes. When succeeding sources have the same shape, the processes converge. A ratio of succeeding sources gives a value to ν_c, the calculated average number of neutrons emitted per fission required to maintain criticality.

The number of groups in a problem refers to the number of values or intervals of energy (or velocity or lethargy) in which the neutrons are assumed to diffuse. The number of mesh points refers to the number of space points at which the flux will be evaluated. The number of regions refers to the number of materials of homogeneous composition which constitute the reactor.

The validity of multigroup calculations is best judged by comparison of the calculated results with experimental observations on actual reactors. The number and scope of such comparisons available so far are inadequate. One may hope that future multigroup calculations will someday have a better experimental foundation.

Computing Machines

Memory. The speed, reliability and cost of digital computers are closely connected with the type of memory. At present, magnetic-core memories are used in the fastest, most reliable, and most expensive machines, such as the IBM 704 and the Remington-Rand 1103A. Cathode-ray-tube memories are cheaper and as fast but have not always proved reliable. The 701, 1103, and other Princeton-type computers all have electrostatic memories, but only MANIAC II is now being built with such a memory. The acoustic mercury-delay-line type of memory is used on the UNIVAC I and the SEAC. This acoustic memory has proved to be reliable and not too expensive, but where speed is concerned, it has the disadvantage of not having random access to the information being circulated as acoustic pulses in the delay line. Magnetic drums are also reliable and relatively cheap, but they do not have the speed advantage of random access.

The size and the speed of the memory usually determine the scope of a calculation. Ordinarily, in discussing the size of the memory, only the number of words* of the "fast" memory is mentioned. Many machines also have auxiliary, or "slow," memories consisting of magnetic drums, tapes, or disks which can store a few million words of information. The UNIVAC I has a 1,000-word memory, and the Princeton-type machines have 1,024- and 2,048-word memories. Magnetic-drum machines have 1,000-, 2,000- and 4,000-word memories. The latest magnetic-core machines, the 704 and 1103A, have 4,000-, 8,000-, 12,000-, and will have 32,000-word memories.

TABLE 1—Commercial Computers

Machines	*Installed*	*On order*
Medium machines		
IBM 650	560	1200
Datatron	55	55
UNIVAC File	10	200
ELECOMS	13	5
ALWAC	16	5
Large machines		
UNIVAC I and II	40	20
UNIVAC Scientific (1103 and 1103A)	20	10
BIZMAC	3	3
IBM 701	16	0
IBM 702	14	0
IBM 704	38	66
IBM 705	46	118

Speed. The speed of a computer can be measured in various ways. A common method is to give the access time, or fundamental machine cycle time. This time interval indicates the time it takes the central processing unit to transmit to or receive from the memory a word of information. Another method is to give the time of a typical multiplication. In a floating-point machine, this time is approximately the same whether the multiplication is carried out by means of fixed point or floating point.† In a fixed-point machine, a floating-point operation may take as much as twenty times as long as the corresponding fixed-point operation. Table 2 lists the number of operations per second performed by various digital machines.

Type. High-speed computers can also be separated according to whether they are one-of-a-kind machines or production models. Until 1956 one-of-a-kind machines, such as Whirlwind I, ORACLE, MANIAC I, ORDVAC, ILIAC, AVIDAC, SEAC, SWAC, NORC, and the MARK I, II, III and IV, played an important role in reactor calculations and, indeed, in all high-speed computing. The present trend is strongly toward renting or buying one of the production models built by a computer manufacturer. With the flood of machines being produced commercially now, it is unlikely that one-of-a-kind machines will again play a major part.

The figures in Table 1 give the approximate number of machines delivered and on order as of early 1957. Including one-of-a-kind machines, there are about 700 medium machines installed and 1,500 on order, and about 200 large machines installed and 250 on order.

Although any of the machines listed

* A *word* is a set of characters, such as 10 alpha-numeric characters or 40 binary digits, i.e., a sequence of 40 0's and 1's, which occupies one storage location and is treated by the computer as a unit.

† In a fixed-point machine, numbers in the machine normally must be between some fixed limits, such as −1 and +1. In a floating-point machine, a number commonly consists of both an exponent and a signed proper fraction.

can be used for technical calculations, some, such as the 702 and 705, were primarily intended for business application. Likewise, the majority of File Computers and IBM 650's undoubtedly will not be used for technical computing.

Future machines will have faster and larger memories and, in the immediate future at least, will cost more. (See p. 60 for more detail.)

Staffing a Computer

The size of the computing group associated with a digital computer seems to be closely proportional to the size of the machine. For example, computing groups associated with the 650 and Datatron usually contain between 3 and 20 people, while groups associated with the 704 and 1103 contain between 12 and 60 people.

Much debate is centered about whether the computing should be carried out in a "closed shop" or an "open shop." In a closed shop, all the programming or coding is carried out by a group directly associated with a digital computer. Problems are presented to this group by the originator, and the solution is returned to him.

In an open shop, the originator of a problem does the coding himself or supervises someone in his group who does it. In the open shop, the personnel associated directly with the machines merely operate them, prepare utility and mathematical routines and perhaps act as consultants on the use of the machine.

An appropriate compromise for groups involved in nuclear calculation is the "service shop," in which the computing group assists the engineer or scientist in whatever fashion is desired, and also carries on certain longer-range problems and programs.

As in most technical professions today, there exists a great shortage of good technical computing personnel. Even more than in most technical fields, including nuclear energy, the expansion in computing during the last five years is astounding. An estimate of 10,000 mathematicians has been made for computing needs in 1958. Longer-range predictions become as high as 150,000 by the late 1960's. Inasmuch as the 1956 combined membership list of the American Mathematical Society, the Mathematical Association of America, and the Society for Industrial and Applied Mathematics contains slightly less than 10,000 names, it becomes shockingly apparent that there will not be enough trained mathematicians to go around.

Even today the majority of computing groups, including those concerned with reactor calculations, make poor use of their machines because they are unable to obtain enough well-trained personnel. It is not uncommon for an individual to waste an amount equal to his monthly salary in a few hours of machine time owing to lack of technical knowledge of either the computer or the problem.

Computer Economics

Use of a digital computer can now be obtained in four ways: purchase, rental, rental with the option to purchase, and the hourly rental of time on a machine at a service bureau. Medium machines rent for $2,000 to $12,000 per month, can be purchased for $40,000 to $300,000 and can be used hourly for $40 to $200. Large machines now rent for about $15,000 to $55,000 per month, can be purchased for $500,000 to $3,000,000 and have an hourly rate of $150 to $750. The monthly rental cost generally refers to a one-shift operation, with each additional shift costing 50% more.

Besides the cost of the machine itself personnel costs represent a major contribution to the total cost of operating a digital computer. A rough rule that applies fairly well to installations renting machines on a one-shift basis is that the personnel costs should be equal to the machine costs. For medium-machine installations, the total costs per year are between $100,000 and $300,000, and for large-machine installations, between $500,000 and $1,300,000.

Cost per Problem

The cost of performing a given nuclear calculation depends on both the character of the problem and the kind of machine used. Since the greatest effort has gone into multigroup calculations, the costs of making these kinds of calculations will be considered in some detail.

One method of estimating the cost of using a high-speed digital computer to solve a multigroup problem is first to estimate the number of "equivalent" multiplicative operations for the problem and then multiply by the dollar cost per multiplicative operation. A rule of thumb for the number of equivalent multiplicative operations is to count the number of multiplications and divisions and multiply by two to account for all other arithmetical and logical operations. The final column in Table 2 gives some representative costs per operation. The following word formula can be used to estimate the number of operations for a multigroup problem:

(source iterations) × (groups) × (mesh points) × (operations per point) × (iterations per group) = operations

Before we give numerical examples, some general comments on the use of this estimate are appropriate:

1. In one dimension, the number of regions and the operations needed for

TABLE 2—Computer Costs

	Operations/sec		*Computer costs*		
Machine	*Fixed point*	*Floating point*	*Rental ($/mo)*	*Purchase ($)*	*Per operation ($)*
Desk calculator	2.5×10^{-2}		5×10^{2}		3×10^{-2}
650, Datatron	10^{2}	15	4×10^{3}	2×10^{5}	6×10^{-5}
UNIVAC	5×10^{2}	10^{2}	2×10^{4}	10^{6}	6×10^{-4}
701, 1103, IAS type	2×10^{3}	10^{2}	2×10^{4}	10^{6}	1.5×10^{-4}
704, 1103A, 709, X-301G	6×10^{3}	6×10^{3}	4×10^{4}	2×10^{6}	10^{-4}
LARC	10^{5}	10^{5}	10^{5}	3.5×10^{6}	10^{-6}
STRETCH	10^{6}	10^{6}	2×10^{5}	6×10^{6}	2×10^{-7}

initially calculating the constant coefficients in the differential equations can increase the estimate by a factor of two. In two and three dimensions, these factors are not significant.

2. The total number of source iterations is the product of the number of source iterations needed to converge to a particular ν_c (the average number of neutrons per fission required to maintain criticality) and the number of ν_c's required to reach the correct value (i.e., 2.5). An adequately converged ν_c usually requires three to fifteen source iterations, four being very common. The number of converged ν_c's required is generally between three and six.

3. The number of group iterations is dependent upon the number of mesh points for the two- and three-dimension problems. This relationship is quite complex and is largely a function of the particular iterative technique used.

4. The final product of source iterations and group iterations can often be reduced (sometimes as much as a factor of 10) by using improved iteration techniques.

The following numerical examples are representative for criticality calculations:

(a) 1-Dimensional 3-Groups; 50-Mesh Points; 3-Regions

$(2)(20)(30)(50)(10)(4) = 2.4 \times 10^6$ op.

(b) 2-Dimensional 30-Groups; 1,000-Mesh Points; 3-Regions

$(20)(30)(1{,}000)(20)(50) = 6 \times 10^8$ op.

(c) 3-Dimensional 10-Groups; 10,000-Mesh Points; 3-Regions

$(20)(10)(10{,}000)(30)(100) = 6 \times 10^9$ op.

The results of these three examples can be combined with the information in Table 2 to give the final costs for the various machines.

In one-dimensional calculations, a complete criticality calculation for sample problem (a) costs, for example, about \$24 on the 704 and \$144 on the 650 or the UNIVAC I. For a single converged ν_c, the amount is about \$6 on the 704 and \$36 on the 650 or the UNIVAC. These figures are confirmed by the running time of the Eyewash code on the UNIVAC and the 704 and the PROD II code on the 650. The computation of a converged ν_c averages about 20 min for 30 groups on the UNIVAC, about 2 min for 32 groups on the 704, and about 40 min for 16 groups on the 650.

In two-dimensional calculations, the cost for a full criticality calculation, such as sample problem (b), becomes almost exorbitant. Even on the 704 or 1103A such a problem would cost \$6,000, while on the 650 or Datatron the cost would be over \$30,000. A single converged ν_c, of course, would reduce the costs to about \$1,200 and \$6,000, respectively. Certainly in these problems one must attempt to minimize the number of groups, mesh points, and iterations. A three-group problem is, for example, not too expensive on the 704 but is expensive on the 650. The running times of a two-group code on the ORACLE, the Mug II code on the UNIVAC, and the Curtiss-Wright code on the 701 bear out these costs.

The 10-group three-dimensional sample problem (c) is very expensive on present machines. If only three groups are used and a single converged ν_c is sought, the 704 price is around \$5,000.

Calculations other than multigroup calculations can also involve a large number of operations. Problems concerned, for example, with burnup, endurance, two- and three-dimensional kinetics, temperature distributions and transport approximations can all involve from 10^6 to 10^{10} operations.

At most computing centers, problems frequently arise which are "one-shot" calculations, i.e., likely to occur only once. Experimental calculations and certain exploratory calculations often fall in this category. The question often arises as to whether hand calculations or a small or medium computer should be used rather than a large computer. The answer is very much a function of the problem, but it is even more a function of the personnel and subroutines available. Given the proper personnel and subroutines, calculations as small as 10^3 operations can properly be placed on large machines.

The time needed to prepare a code is largely dependent upon the individual programmer and almost independent of the type of machine involved. The time to "debug," i.e., verify that the code is running correctly, is likewise strongly dependent upon the individual programmer and, to a much lesser extent, on the particular machine in use.

In summary, the cost of coding and debugging a given program is almost independent of machine size, while the cost per mathematical operation is much smaller for larger machines (see Table 2). The size and the repetitive nature of most reactor calculations therefore indicate the use of the fastest (and therefore generally cheapest per problem) commonly available computer. A commonly available computer is desirable because of the economic advantage of interchanging codes with other installations which are working on similar problems.

Digital Computers of the Future

The two largest computer manufacturers, Sperry-Rand and IBM, have each announced two new large machines. Sperry-Rand is working on the X301-G and the Livermore Automatic Research Computer (LARC), while IBM is working on the 709 and the STRETCH.

The X301-G and 709 will not operate significantly faster than the present 1103A and 704, but because of larger memories, more versatile orders, and advantageous input and output equipment, many problems will probably be computed in half the time now required.

The LARC is expected to be faster then the present 1103A and 704 by a factor of about 15. It will read data from its ferrite core memory at a rate of 2 μsec per word. The initial capacity of the fast memory will be 20,000 ten-digit numbers. This may be substantially increased by adding supplementary fast memory units. A LARC is to be delivered to Livermore in early 1958, while additional LARC's are on order for a year or two later. The contract price for the construction of the computer is \$2,895,000.

IBM's STRETCH, in turn, is expected to be about a factor of 10 faster than the LARC. It will also have a magnetic-core-type fast memory. A data word can be read from the memory within 0.2 sec. The machine is being designed to accommodate a fast memory capacity of up to a million words. In addition, external memories (magnetic discs and magnetic tapes) may ultimately provide data in blocks up to a total capacity of 100 million words. STRETCH is being built for Los Alamos and is to be delivered about January, 1960.

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Control Analysis with a Digital Computer

In the past, systems analysis of reactor control was reserved for real-time analog computers. Now digital computers threaten to take over part of this critical nuclear-design function

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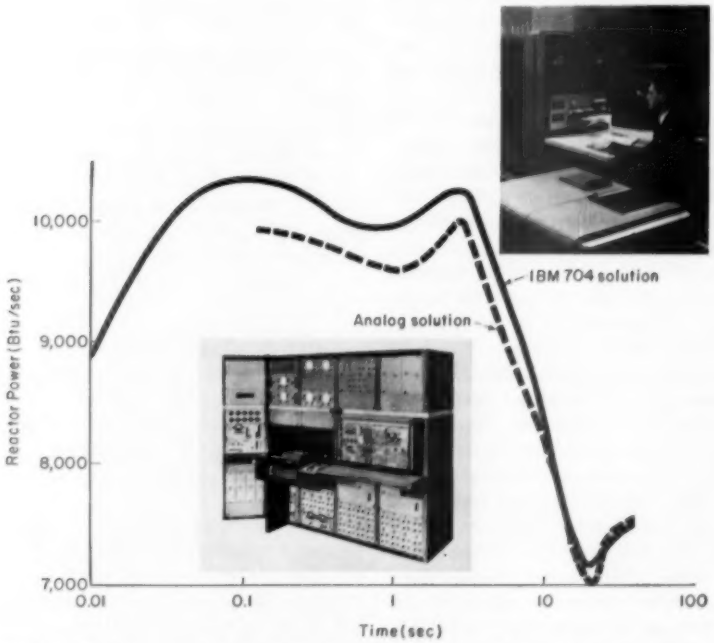

FIG. 1. Comparison of digital computer and analog solutions of reactor control problem. Shown is the time behavior of the reactor power following a 0.002 increase in reactivity

THE ANALYSIS of reactor control is perhaps one of the most complex of all the calculations made in designing a reactor. The aim of such a study is to predict the transient response of the reactor system to various hypothetical situations. The analysis therefore involves the time-dependent interrelationships of all the major components in the reactor plant. Traditionally this type of problem is considered only for the analog computer. However, for our study of the control characteristics of the Martin Power Reactor (MPR) we also coded the problem for a digital computer. We were then able to compare the merits of the two techniques by making parallel studies on the analog and digital devices. The results are shown in Fig. 1. Our conclusion is that the digital approach gives results that are as good as the analog results and that in addition it has certain advantages.

Analog vs Digital

The fundamental advantage of the analog is that it simulates the problem by physically representing it in the form of an electrical system. Because of this, the engineer has a better physical feeling for the control system. Changes of input conditions are easily taken care of because normally they require only the resetting of a potentiometer. Also the analog computer is capable of "real-time" solutions—the analog can run on the actual time scale of the physical situation being simulated.

The analog has its disadvantages as well as advantages. For example, an extensive check-out procedure is required whenever new parametric values within the problem are investigated, even though the patch boards may have been saved. Also, extreme care must be taken to prevent output voltages from becoming so small that the noise of the system enters into the results. This possibility arises in working with equations that have terms composed of large and small factors (as in the problem under discussion). The relative voltage levels representing the factors must be scaled in such a way that none of the voltages becomes so small that system noise introduces error.

Another difficulty lies in the choice of recorded variables and recording levels to prevent loss of detail. For example, on certain runs the choice of a particular control parameter caused

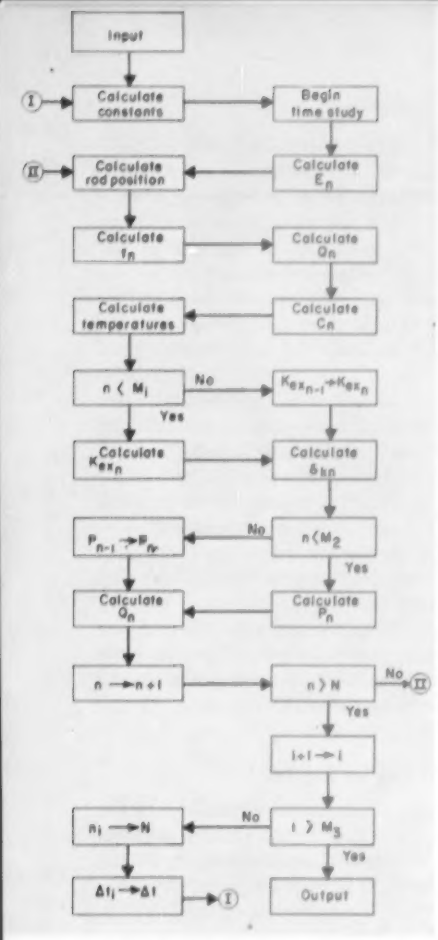

FIG. 2. Logical flow chart for digital-computer calculation. Subscript n refers to particular point in time. M_1, M_2, M_3 and n_i are numbers that specify (in order) (a) time interval for ramp variation of induced reactivity, (b) the time interval for ramp variation of reactor output load, (c) the number of changes in the incremental time step, Δt, and (d) the length of time for which a given Δt_i is used

the control-drive mechanism to alternate the direction of drive rapidly. Since only rod position was recorded this chattering was not detected on the analog run. It was, however, immediately apparent in the IBM 704 digital output for the rod position. (A more careful choice of variables, such as rod speed in this case, would have shown the effect on the analog.)

Because of these disadvantages and uncertainties, it was felt that a solution obtained by an independent method would be desirable. This led to the programming of the same problem on a digital computer (IBM 704). The primary objectives were to (a) check the analog solution and (b) investigate the feasibility of digital techniques in this area.

Although the results are somewhat incomplete at this time, it is already evident that the digital approach offers important advantages in some respects. For example, the reinvestigation of a particular problem at a future date involves a comparatively extensive set-up procedure for the analog computer. The digital computer, however, requires only the preparation of a few more punched cards for input data.

The Problem

The MPR control system is of the on-off type—control rods move only when the actuating signal is above a certain level. The central analytical problem was to specify a control system that would tolerate small variations in reactor power and temperature without causing control-rod motion but would still be sensitive enough to satisfy operational requirements. The analysis involves a set of differential equations relating the control system to the primary coolant loop and the neutronics of the reactor. Differential equations of this sort can be readily solved by digital-computer techniques.

The time behavior of the important physical variables of the reactor system is described analytically by the set of equations tabulated below.

The equations used to represent the primary loop consist of eight representing the reactor kinetics, seven covering the heat transfer in the loop, two for transport of the coolant and two defining algebraic equations.

Equation 1 is symbolic of each of the six delayed-neutron equations. Equation 2 represents the rate of heat production in the reactor core due to the change in reactivity as defined by Equation 3.

Equations 4 and 5 represent the heat transfer in the reactor core while Eqs. 9, 11 and 12 show the same thing in the heat exchanger. Equations 6 and 10 are the defining equations for the mean coolant temperature in the reactor core and heat-exchanger tubes. Equations 7 and 13 are general mixing functions for the upper section of the reactor and the outlet header of the heat exchanger.

Equations 8 and 14 introduce a constant time delay between reactor outlet and heat-exchanger inlet temperatures and between heat-exchanger outlet and reactor inlet temperatures.

$$\frac{dC_i}{dt} = -\lambda_i C_i + \frac{\beta_i}{l^* k^*} Q \qquad (1)$$

$$\frac{dQ}{dt} = \frac{1-\beta}{l^*}\delta k Q - \frac{\beta}{l^*} Q + k^* \sum_{i=1}^{6} \lambda_i C_i \qquad (2)$$

$$\delta k = k_{ex} + R_w \left[\frac{X - X_o}{H} - \frac{1}{2\pi}\left(\sin\frac{2\pi X}{H} - \sin\frac{2\pi X_o}{H}\right)\right] + \alpha \Delta T_w \qquad (3)$$

$$\rho_m C_m (r_o - r_i)\frac{dT_m}{dt} = 2h_r(T_w - T_m) + \frac{r_o - r_i}{v} Q \qquad (4)$$

$$r_i \rho_w C_w \frac{dT_w}{dt} = 2h_r(T_m - T_w) + \frac{r_i \rho_w C_w u_r}{H}(T_a - T_i) \qquad (5)$$

$$T_w = 0.497 T_i + 0.503 T_o \qquad (6)$$

$$\frac{dT_{ro}}{dt} = \frac{R}{V_{ro}}(T_o - T_{ro}) \qquad (7)$$

$$T_{bi}(t) = T_{ro}(t - d_1) \qquad (8)$$

$$r_{ib}\rho_w C_w \frac{dT_b}{dt} = \frac{r_{ib}\rho_w C_w u_b}{W}(T_{bi} - T_{bo}) - 2h_i(T_b - \theta_m) \qquad (9)$$

$$T_b = \tfrac{1}{2}(T_{bi} + T_{bo}) \qquad (10)$$

$$\rho_{mb} C_{mb} \frac{r_{ob}^2 - r_{ib}^2}{r_{ib}} \frac{d\theta_m}{dt} = 2h_i(T_b - \theta_m) - \frac{2h_o r_{ob}}{r_{ib}}(\theta_m - T_s) \qquad (11)$$

$$M_s C_s \frac{dT_s}{dt} = 2\pi h_o r_{ob} n W(\theta_m - T_s) - P \qquad (12)$$

$$\frac{dT_{go}}{dt} = \frac{R}{V_{go}}(T_{bo} - T_{go}) \qquad (13)$$

$$T_i(t) = T_{go}(t - d_2) \qquad (14)$$

$$E = \tfrac{1}{2}(T_{ro} + T_{go}) - 970 + D\frac{dQ}{dt}\Big/ Q \qquad (15)$$

To simulate the on-off control system the signal actuating the control-rod drive is given by Eq. 15, where 970 is the desired mean temperature of the reactor core in °R and $(dQ/dt)1/Q$ represents the voltage to the period meter.

When the absolute value of the signal reaches a predetermined level, it actuates the control-rod drive, which operates until the absolute value of the signal decreases to the desired level. The start and stop levels can be set separately so optimum operation of the dead zone can be determined.

Digital Solution

When a set of differential equations are to be solved on a digital computer, such as the IBM 704, one of the most common approaches to the problem involves the replacement of the differential equations by a set of finite difference equations. These finite difference equations then relate the values of the dependent variables at some discreet point in time to the values at some set of previous times. The method of solution is then to step the solutions along in time by starting from a known set of initial conditions.

The equations of basic importance in the solution of this problem are Eqs. 1 and 2. For Eq. 2, the most convenient difference representation is one involving only first derivatives. Previous experience on a preliminary study of this equation using the IBM 701 led to a choice of the simplest form for the time dependence of the reactor power, Q:

$$Q(t_{n+1}) = Q(t_n) + \Delta t \frac{dQ}{dt}(t_n)$$

At every step in the calculation truncation and round-off errors are introduced. These errors are propagated from step to step, and their growth may cause them to dominate the solutions of the equations. This numerical instability of the difference equations is a major obstacle in a problem of this type.

Although an analysis of the numerical stability characteristics of this approximation is possible, it is extremely difficult to obtain estimates that are not excessively conservative.* However, preliminary calculations on an IBM 701 computer gave some practical experience in handling a similar set of equations and indicated that the computation time would not be excessive.

The preliminary investigations also indicated the desirability of being able to change the incremental time step, Δt, at various stages of the calculation. Thus, in the case of a step input of reactivity, a 1-msec time step is used for the first few hundred milliseconds. The Δt is then changed to 10 msec.

The preliminary investigation also indicated some difficulty with the precursor equations (Eq. 1). An unfortunate choice of numerical approximation to the quantity $e^{-\lambda_i \Delta t}$ affected the stability of the system of equations. The following method was used on the 704 computer in order to avoid this problem. The solution of the precursor equation can be written

$$C_i(t) = C_i(0)e^{-\lambda_i t} + \frac{\beta_i}{l^*} e^{-\lambda_i t} \int_0^t Q(s) e^{\lambda_i s}\, ds$$

The difference equation is obtained simply by calculating the quantity

$$C_i(t_{n+1}) - e^{-\lambda_i \Delta t} C_i(t_n)$$

and approximating the integral by the trapezoidal rule:

$$C_{i\,n+1} = e^{-\lambda_i \Delta t} C_{i\,n} + \frac{\beta_i \Delta t}{2l^*} [Q_{n+1} + e^{-\lambda_i \Delta t} Q_n]$$

The remaining equations are treated like the precursor equations except where iteration would be required, in which case the treatment is the same as for Eq. 2. The time-delay equations are handled simply by retaining values of the necessary variables and picking out the value at the proper time. Interpolation is possible so that the choice of time step can be completely arbitrary.

Once the problem was coded, checkout and investigation of stability and truncation error proceeded more or less simultaneously. Although each new type of problem should be investigated separately, the general approach is to use a very small time step during any sharp transients and then to increase the time steps during the periods when the transients are not sharp.

Figure 2 shows the logical flow diagram for the 704 calculation. Many of the details have been left out in the interest of simplicity. Nevertheless, the order of calculation and the essential features of the program are given. Many of the boxes shown represent complex logical or arithmetic operations. For example, in the calculation of control-rod position control-rod speed and position must be computed according to a complicated function of time, subject to the control of the actuating signal E.

The machine program is capable of handling a wide variety of problems. There are provisions for both step and ramp changes of reactivity and load. All constants and initial conditions were utilized as input data so that the initial operating state that existed when the transient started could be simulated. Twelve of the dependent variables have been recorded, and they may be printed out at every step, or an arbitrary number of steps may be passed over between recorded values.

Analog Solution

The analog problem was handled by four Reeves "400" series machines consisting of 103 operational amplifiers, two electronic multipliers, one polar-to-rectangular-resolution resolver, 100 attenuators, 2 six-channel recorders and associated equipment. Control-rod drive was simulated by three relays that switched the accelerations on and off as required by the control signal. The time delays were handled by building up a block of equipment whose transfer function was approximately e^{-sd}. Therefore, if $F(t)$ was fed into the block, the output was approximately $F(t - d)$. The simulation of the time delays themselves completely utilized one of the "400" series machines. The logic of the analog problem is identical to that of the 704 code.

Results

The results of these studies were used to pick the most desirable control-rod speed and to determine other parameters not previously set by design. Using values for control-rod speed and other parameters thus determined, the Martin Power Reactor showed very good stability for all cases of step and ramp changes in reactivity and step changes in the power requirements of the secondary system. The time variation of the power, Q, is shown in Fig. 1, as a sample of the results of a step in reactivity as determined by analog and by digital computers. The agreement is probably better than the accuracy of the equations themselves.

It is therefore feasible to solve the control problem with either type of computer. The most desirable strategy, however, is to employ both computers in a study of this kind. This gives a check on the set-up of each of the computers. In addition each machine can then be used in the phase of the study for which it is best suited. The analog computer, because of its flexibility, can be used most conveniently for early, exploratory analysis. Then after the completion of the analog run, corrections and refinements can easily be made with the digital program and the solution rechecked in a relatively short time.

* F. B. Hildebrand, "Introduction to Numerical Analysis," p. 222 (McGraw-Hill Book Co., Inc., New York, 1956).

Monte Carlo Solves Reactor Problems

The Monte Carlo method handles neutron-transport problems by tracing simulated neutron histories. Faster computing machines and programing ingenuity will encourage future use in nuclear calculations

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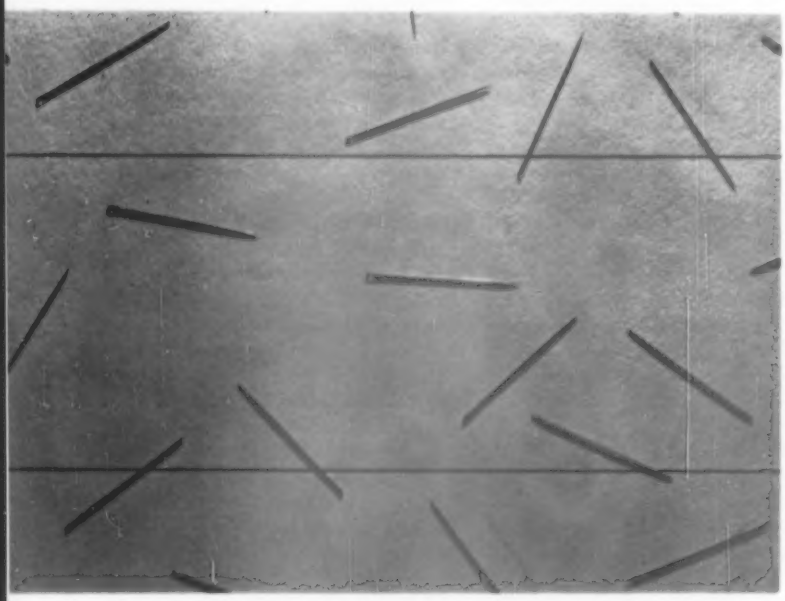

FIG. 1. Needle problem introduced by Buffon (1707–1788) was first Monte Carlo calculation. Needle is thrown at random on surface marked with parallel lines spaced two needle-lengths apart. Total throws divided by times needle hits lines approaches π as number of trials is increased

THE MONTE CARLO method is a technique for solving physical and mathematical problems by using random sampling of an analog nature. Although the method can be used for many kinds of calculations, including some that have no apparent relation to statistics (See Fig. 1), the most extensive and successful applications have been to physical problems that are inherently statistical.

In particular the Monte Carlo approach provides straightforward solutions to many difficult problems in reactor physics. Monte Carlo programs for digital computers are being used now to solve problems of shielding, neutron slowing down and resonance escape and may eventually be used for complete reactivity calculations.

The problem of neutron or radiation transport is basically concerned with random events—the most that can be stated about the interactions of neutrons with matter is the probabilities per unit length (i.e., the macroscopic cross sections) for various kinds of events—scattering, absorption, fission, etc. In many of these events the outcome (e.g., the possible directions of neutron scattering, and energies of fission neutrons) can be specified only in terms of probability distributions.

In a Monte Carlo calculation of neutron transport, these stochastic processes are simulated, by hand or on an automatic computer, by following "histories," i.e., trajectories, that have the same alternatives, with (usually) the same probabilities as real neutrons. Random numbers are used to decide whether a "neutron" is absorbed, and a history terminated, or scattered and a new direction and energy chosen in accord with the scattering law and collision mechanics.

If an automatic computer is used the random numbers can be either fed into the machine as input or generated within the machine itself by an auxiliary program. The most usual set is a sequence of numbers, R, which lie anywhere in the range 0 to 1 with equal probability. Such a sequence of random digits can be obtained from tables like that compiled by the Rand Corporation (*1*).

The selection of one or more random digits then determines the outcome of each neutron event. For instance, the distance between collisions (measured in units of the mean free path) has the probability distribution e^{-s}. If R is

the selected random number, then the distance to the next collision is given by $x = -\ln R$. Similarly, to choose the cosine, ω, of a scattering angle in an isotropic scattering event, put $\omega = 2R - 1$.

In this way it is possible to simulate exactly (insofar as the nuclear cross sections are known) the behavior of a neutron in a chain-reacting assembly. By following enough histories, it is possible to determine the average behavior of neutrons—to count the number of fissions, of parasitic absorptions, of escaping neutrons, and so on.

The Monte Carlo result, however, is exactly correct only when an infinite number of independent estimates of a given result have been obtained. Any answer that derives from a finite number of tries has a statistical dispersion or error.* In general this error depends upon the number N of independent histories calculated, decreasing when N is large as $N^{-1/2}$. The statistical error can therefore always be reduced by using more computing time (generating more histories). In addition the accuracy that can be obtained for a fixed amount of computing time can often be increased by using appropriate short cuts. For instance, it is not necessary, as we shall see, for a Monte Carlo calculation to follow all the possible fluctuations available to a neutron; generally when it is possible to use the average value of some variable rather than to choose at random among alternative values, the accuracy obtained for a given effort increases.

Advantages. The Monte Carlo method can simulate, at least in principle, any neutron interactions that are physically understood. Monte Carlo procedure is equivalent to obtaining an exact solution of the Boltzmann transport equation; therefore, physical situations not accessible to the usual analytical approximations present no particular problems. For instance, Monte Carlo can treat regions of space with dimensions small compared with a neutron mean free path—a situation where diffusion theory can not be used. Rapidly varying nuclear cross sections and anisotropic neutron scattering can also be dealt with.

In addition, the fact that Monte Carlo calculations are usually closely analogous to physical processes means that the setting up of calculations is often a simple extension of understanding of the physics of a situation. The programing and coding of Monte Carlo methods for an automatic computer can therefore be simpler than the corresponding work in diffusion or transport theory. The physical analogy can work the other way, too; the detailed observation of the histories can help the development of approximations and physical pictures which can advance a more analytical approach.

Applications. The particular advantages of Monte Carlo have been exploited in a number of calculations relating to the design or understanding of portions of reactors. For example, Mayne, Pendlebury and Woodcock (*2*) have used random sampling of the slowing down of neutrons in an infinite medium to calculate multigroup constants for later use in an analytical treatment. Coveyou, Bate and Osborne (*3*) have used Monte Carlo to investigate the effect of thermal motion of moderator and absorbing atoms upon the thermal and epithermal neutron flux spectra in an infinite medium. Richtmyer (*4*) and Sampson (*4*) have computed resonance-capture probabilities in water- and graphite-moderated lattices. In the Richtmyer work, resonance scattering and absorption and Doppler broadening are treated.

Although these calculations are all quite complicated, they are each concerned with only a portion of the reactor problem. In principle it is possible to do the complete reactor criticality calculation by the Monte Carlo technique. For a fixed neutron source, we can follow a group of histories to their conclusions and count the number of fissions produced. These fissions can then be used to generate a new source so that iterations to a final power pattern may be carried out.

So far Monte Carlo programs that make complete reactivity calculations are not available to the reactor designer as practical tools. Several groups, however, are actively working toward this goal, and Leshan (*5*), for instance, has coded a general-purpose Monte Carlo program for the 704 that might be used for this task.

Disadvantages. For Monte Carlo studies in general the problem of getting enough accuracy (or of getting enough time on an automatic computer to achieve it) is often a serious drawback. This becomes more and more serious as the required information about the system becomes more and more detailed.

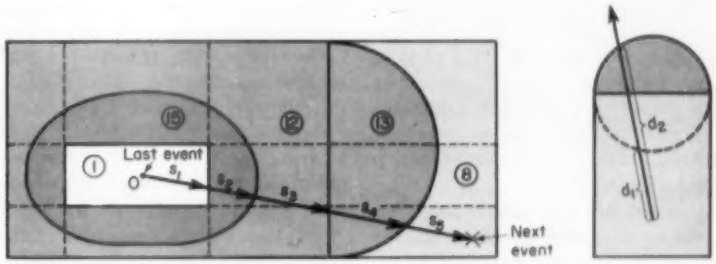

FIG. 2. Logical problems in Monte Carlo neutron-transport calculation. Left: Given total distance and direction of next event with respect to last event, automatic computer must identify region containing new event. Right: Computer must recognize whether bounding surfaces are real (solid circle) or fictitious (dotted circle)

If the calculation is expected to yield information concerning the neutron flux as a function of both space and energy, it becomes difficult to achieve statistical reliability in a reasonable time on present-day machines. Even without this, for a system in which the neutron is expected to make many collisions before it is absorbed or escapes, the calculation may be very slow. But this is exactly the situation where diffusion theories are most reliable. Thus statistical and nonstatistical methods can often complement each other. A good example of this is the Monte Carlo calculation of multigroup constants mentioned above.

Generally speaking, analytical or nonstatistical numerical methods, where adequate, tend to be more economical than Monte Carlo methods; exceptions may occur when a problem requires a

* This error is usually stated in the form of a limit ϵ such that the true answer can be expected to lie within the experimental answer plus or minus ϵ a certain fraction of the time. For example, the result of a certain statistical experiment might be stated as 26 ± 4 to "70% confidence." This means that if the experiment were repeated many times, then the fraction of the time that the result lay between 22 and 30 would approach 0.7.

"quick and dirty" answer and the direct and intuitive appeal of the statistical method is useful. At the moment, therefore, the use of Monte Carlo is limited to the reactor designer who has plenty of machine time and who wants to investigate physical effects or geometries too complicated for the standard analytical methods. We might expect that in the future Monte Carlo applications will become much more common as experience is acquired and computing machines become faster.

A Monte Carlo Code

The best way to get a feeling for what is involved in the Monte Carlo technique is to consider a particular calculation in some detail. Perhaps the ultimate in Monte Carlo calculations is an automatic-computer program that will calculate the reactivity of a reactor. We will therefore describe at some length the significant features of a complete Monte Carlo reactivity code.

A Monte Carlo reactor calculation is carried out by following a large number of hypothetical neutrons from birth to death whose behavior is governed at each point by probability distributions derived from physical laws. For each neutron, the sequence of events is called a "history."

A particle history begins by assigning to the neutron an energy from the fission spectrum and a direction of motion.

The distance (measured in mean free paths) to the next event is selected from an exponential distribution. If this is outside the reactor, the particle has escaped; otherwise we choose a collision type. If this is an absorption, the particle is finished and we choose a new neutron and follow it. If it is a scattering collision, we choose a new energy and direction and go on to the next event. We continue to follow the neutron until it is absorbed or escapes.

This is done for all of the original source particles. Neutrons created by fissions of these source particles become the source for the next generation, and so on.

In the following paragraphs we shall discuss in somewhat greater detail each of these parts of the code.

Birth. To give birth to a neutron one must specify six coordinates: three of position, two of direction of travel, and an energy. The position may be selected, initially, at random in the fuel region of the reactor. After a complete generation, however, neutrons of the next generation will be born at the points where their parent particles made fissions.

The direction of travel can be chosen at random from a distribution that is isotropic in space. Thus if w_x, w_y, w_z denote the direction cosines of the velocity vector, we may choose

$$w_x = \sin\varphi \sin\theta$$
$$w_y = \cos\varphi \sin\theta$$
$$w_z = \cos\theta$$

where Prob $\{\varphi\}d\varphi = 1/2\pi\, d\varphi$
and Prob $\{\cos\theta\}d(\cos\theta) = d(\cos\theta)$.

In the absence of external sources the energy is chosen at random from the fission spectrum.

Flight path. The number of mean free paths to be traveled by the neutron must come from the distribution Prob $\{X\}dX = e^{-X}\, dX$, where X = number of mean free paths.

One way of choosing such an X is to take $X = -\ln R$, where R is a random number. A method that can be more efficient, and is certainly more elegant, is given in ref (*6*).

Geometry. The function of the geometry part of the code is to solve the following problems:

Given a neutron at a point in space traveling in a known direction, and given also the fact that it will travel exactly X mean free paths, then

- What regions of the reactor will it pass through?
- What will be the length of the path segment in each region?
- What will its coordinates be at the end of the path?
- In what region will it be at the end of the path?

These are all questions of no conceptual difficulty whatever. However, the method used to automatize their solution on a computer may result in tremendous time savings or losses per history. The geometrical part is, therefore, the heart of the Monte Carlo program, and the greatest care and ingenuity should be devoted to this portion of the code.

Figure 2 shows the typical situation. The neutron is born in region 1, moves through regions 15, 12, 13 and finally makes a collision in region 8. We have

$$X = \mu_1 s_1 + \mu_{15} s_2 + \mu_{12} s_3 + \mu_{13} s_4 + \mu_8 s_5$$

where the μ's are macroscopic total cross sections at the neutron energy in each region.

In the calculation of s_1, for example, one finds the intersections of the neutron trajectory with each of the geometrical surfaces bounding region 1. One then calculates the distance from the starting point to each point of intersection, rejects "irrelevant"(see below) intersections and those corresponding to the neutron traveling in the wrong direction, and takes the minimum of the remaining distances as s_1. This process is repeated in regions 15, 12, 13, 8 until X is exhausted. It is useful to tabulate, for each bounding surface, the regions that lie on each side of it, so that one can find easily that region 15 was crossed into from region 1. The alternative of identifying the region from the coordinates of the intersection is too horrible to contemplate.

Figure 2 shows a spherical header on a cylindrical region. One might easily make the error of calculating the distance d_1 as the nearest intersection with a bounding surface if one just has the equation of the spherical cap. Clearly d_2 is the one desired. d_1 corresponds to an irrelevant intersection. Input data to the code must therefore include the information that the hemisphere lying below the upper plane of the cylinder is not a physical surface and that intersections with it are to be disregarded.

Concerning the geometric specification of the surface elements, a useful and very general prescription, due to Leshan (*5*), is to prescribe the constants $a_1, \ldots, a_{10}$ of the quadric surface

$$P(x, y, z) = a_1x^2 + a_2xy + a_3xz + a_4y^2 + \cdots + a_7x + a_8y + a_9z + a_{10}$$

This form allows great freedom in choosing the shapes of bounding surfaces, but may result in many unnecessary operations if, for example, only cylinders whose axes are parallel occur in the reactor. In the latter case, one would prescribe only the radius, height and coordinates of the center for each such cylinder.

Sturm's theorem (*7*) for the isolation of the roots of a polynomial equation is a useful device for rejecting a surface which is too far away (i.e., further than the nearest relevant intersection so far found). The application of this test can avoid many square-root operations.

Neutron event. At the end of the path of flight, the neutron can cause a fission, be absorbed or slow down elastically or inelastically. The fate of a given neutron is decided as follows:

Let R be a random number between zero and unity. If $R < \mu_a/\mu_t$ the par-

ticle has been absorbed. If not, but $R < (\mu_a + \mu_f)/\mu_t$ the particle has made a fission. Failing this, if

$$R < \frac{\mu_a + \mu_f + \mu_{in}}{\mu_t}$$

an inelastic scatter has occurred. Otherwise the neutron has been elastically scattered. To determine by which element it has been scattered let μ_{si} denote the scattering cross section of the ith element of the region at the neutron energy, and let $\mu_s = \sum_i \mu_{si}$. Determine i (and therefore the element responsible for the elastic collision) from the condition

$$\frac{1}{\mu_s}\sum_{j=1}^{i-1} \mu_{sj} \leqq R' < \frac{1}{\mu_s}\sum_{j=1}^{i} \mu_{sj}$$

where R' is a new random number.

Knowing the atomic number of the element and the scattering angle chosen isotropically (for example), one can then deduce the new energy from the angle-energy correlation law.

In case of an inelastic collision the new energy must be obtained from tabulated distributions.

If an absorption has occurred one simply records that fact and goes to the next particle.

In case of a fission one might expect to play a game of chance resulting in the creation of an average of 2.5 new particles for the next generation. We suggest, however, that a fission be treated precisely as an absorption and that new particles be created at *each collision* by the following method:

Let $\nu\mu_f/\mu_t = I + X$, where I is an integer and $0 \leqq X < 1$ and ν is the average number of particles produced per fission. Then at each collision produce I particles and choose a random number R. If $R \leqq X$, then create an additional particle. This procedure spreads out the fission source over space more rapidly than simply producing two or three particles at each fission and thus reduces the number of histories required to determine the spatial distribution of the fissions.

Estimating k. The reactivity may be estimated in many ways, some of which are as follows:

(a) The ratio of the number of particles in generation $n + 1$ to the number of particles in generation n.

(b) The ratio after many generations of 2.5 to the number of parent neutrons per neutron going into fission.

(c) Same as (b) but using statistical estimates of the number of neutrons going into fission. Such an estimate may be made at each collision simply by accumulating the ratio of fission to total cross sections in the region at the collision energy.

Of the three methods (c) gives the most accurate result for a given number of histories.

Refinements

In some situations the calculation can be simplified or speeded up by using Monte Carlo procedures that do not correspond to the actual physical train of events. Such deviations from reality, of course, are justified provided they give the correct results.

The useful kinds of changes from a direct Monte Carlo analog depend upon the kind of problem being done.

A good example of a situation where these techniques are very helpful occurs in calculating the penetration of neutrons or gamma rays through thick shields. Since the accuracy of the results depends upon the number of emergent particles, attention must be focused on particles that are most likely to achieve this final success. For example, if a slab shield is six mean free paths thick, then about 0.0025 of the source particles survive without making collisions. To get an estimate of the penetration accurate to about 10%, we need about 100 emerging particles. In a straightforward Monte Carlo calculation about 40,000 histories would then be necessary.

Since a significant amount of machine time would be required to generate this many histories, techniques are used to improve the calculating efficiency.

One of the simplest of these techniques is a device that is usually known as "splitting." The slab is partitioned into a number of regions by a set of planes parallel with the slab. Each neutron is assigned a weight w that multiplies the neutron's contribution to the final result. During the neutron's history its weight is changed as follows: When a neutron traveling in the direction of escape crosses a dividing plane, it is split into two neutrons each with a weight one-half of that of the original neutron. When a neutron crosses a plane traveling in the opposite direction, its weight is doubled. The neutron is then forced to play a game of "Russian Roulette" that allows it a 50% chance of survival. With this arrangement most of the time is spent following the particles that will most likely escape. The use of neutron weights removes bias in the final tally. Splitting can thus be an effective method of improving the efficiency of a shielding calculation (*8*).

In general, techniques to improve efficiency involve either a modification of the probability distributions used to determine the neutron's fate or a change in statistical estimation (i.e. the scoring procedure).

Probability modification. Suppose that at a certain step in a Monte Carlo history a particular event (e.g., fission, or elastic scattering into a specific angular range) has probability p. It is possible to generate the history under the assumption that the event has a probability p' provided we associate with this history a weight w (initially set equal to 1) which is multiplied by the ratio p/p' after this event. Every score derived from this history must be weighted with the current value of w. For example, if at a collision the expected number of fissions is estimated as described above as μ_f/μ_t, then this is to be multiplied by the weight w of the history before the collision. It can be shown (*8, 9*) that a proper choice of altered probability distributions for each step can often increase greatly the efficiency of a calculation.

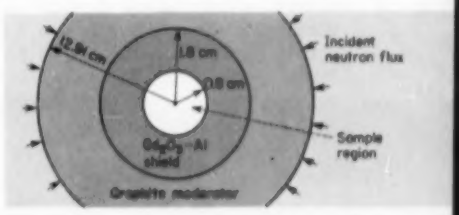

FIG. 3. NDA Monte Carlo program calculates fraction of incident neutrons that reach sample region

When the change in the probability distribution serves to focus attention on certain kinds of histories the method is called "importance sampling."

A useful application of the device of choosing from a wrong probability distribution is the suppression of absorptions. At a collision where scattering and absorption can occur, with the probability of the former being μ_s/μ_t, the history always proceeds via scattering and the weight of the history is reduced by the factor μ_s/μ_t. In this case, histories are terminated only when par-

ticles leak. It may be desirable also to terminate a history when w, the weight, becomes very small. In a reactivity calculation, a game of chance should be played at each collision in a fuel region to determine whether fission neutrons are born. These neutrons must then be assigned their parent's weight at birth.

Another, and perhaps even more important, application is in the estimation of small changes in the properties of a system when some parameter is varied. Briefly, the device is to calculate the histories using one value of the parameter. For a different value of the parameter, these histories are chosen from the wrong distribution, but can be used if we attach the proper weights to them. We can now get two (or more) sets of scores which correspond to different values of the parameters. Since the same statistical fluctuations appear in all, the differences in the answers are much more reliable than if they had been computed independently. The whole set of answers will generally take less time, too.

Summary of Sample Problem Results

Cosine of incidence angle	*Particles starting**	*Particles entering sample*	*Limits for 70% confidence*
0.948	725	608	±25
0.835	500	33	± 5.7
0.704	500	30	± 5.5
0.540	500	20	± 4.5
0.298	500	13	± 3.6
TOTALS	2,725	704	±26.5

* The particles were selected from a $1/E$ neutron spectrum with an upper energy limit of 30 ev.

As an example suppose we wish to estimate the effect of changing the concentration of one element in a region in which each material is capable only of scattering and absorbing neutrons. The change is reflected in the total cross section and in different values for the probabilities for each kind of interaction. Suppose the histories are generated for the original values of the parameters $\mu_t^{(1)}$, etc. Two sets of weights are kept: w_1, w_2. Now the probability that a collision occurs at a distance between s and $s + ds$ (with the entire path in one region) is $e^{-\mu_t s} d(\mu_t s)$. Thus for a collision in this region w_2 is multiplied by the factor $\mu_t^{(2)}/\mu_t^{(1)} \exp[-(\mu_t^{(2)} - \mu_t^{(1)})s]$ and w_1 is unchanged. If the collision occurs in another region and s is the length of path through the altered region, w_2 is multiplied simply by $\exp[-(\mu_t^{(2)} - \mu_t^{(1)})s]$. At a collision in the changed region (in which absorptions are suppressed, as above) w_1 is reduced by $\mu_s^{(1)}/\mu_t^{(1)}$, w_2 by $\mu_s^{(2)}/\mu_t^{(2)}$. When a scattering is made in element i, w_1 is unchanged, w_2 multiplied by

$$\frac{\mu_{s,i}{}^{(2)}\mu_s{}^{(1)}}{\mu_{s,i}{}^{(1)}\mu_s{}^{(2)}}$$

It is important that the comparison does not introduce new alternatives.

Statistical estimation. By statistical estimation we mean any change in scoring for some event so that we no longer merely count the number occurring in a history. The use of μ_f/μ_t at each collision as an estimate of fissions, as described above, is an example of statistical estimation.

A more sophisticated kind is often used in penetration problems. Following each collision, after a new direction has been chosen, but before we determine the position of the next collision, we can estimate analytically the probability that the particle coming out of the collision will emerge from the shield (this is just e^{-X}, where X is the number of mean free paths to the outside measured along the path). In this way, each particle contributes to the estimate of escapes. In some simple geometries (e.g., point detector) it is desirable or possible to estimate the expected contribution *before* a particular direction is chosen for the new path.

There is another, very simple kind of statistical estimation that costs almost no additional time. The sum of all the track lengths (s) for particles in a given region divided by the volume of the region and the number of particles is an estimate of particle flux. Hence $\mu_f s$ and $\mu_a s$ can be used as estimates of the expected number of fissions and absorptions in a region. The sum of such quantities taken over all histories should yield a more reliable estimate of k than the numbers of fissions and absorptions counted up in the histories.

Sample Transmission Problem

A Monte Carlo program coded for the NDA Datatron computer was used to calculate the energy spectrum of neutrons incident on a sample enclosed in a small spherical shield which, in turn, is imbedded in the moderator of a graphite-moderated reactor (see Fig. 3). The code solves the problem of neutron transport in several concentric spherical regions of different compositions where the neutrons come from a distributed source.

Inelastic scattering, anisotropic scattering and splitting are all included in the problem. The code prints as output the neutron flux (calculated from track lengths) in prescribed spherical shells, and the inward and outward current across the surfaces of these shells, each in prescribed energy ranges.

Diffusion theory was inapplicable here for two reasons: first, the incident neutron energy spectrum was approximately $1/E$, and the shield was a strong absorber with a strongly energy-dependent absorption cross section; second, some neutrons might make multiple traversals of the graphite-shield interface. (The calculations later showed the latter was unimportant.)

In addition to the geometric specifications and compositions, the angular distribution and energy spectrum of the incident neutrons were input data.

The results of the calculation are summarized in the table.

The Datatron running time for the problem was about 90 min, or approximately 6 sec per history. Of the original 2,750 particles, 25.6% ultimately reached the sample. At about 70% confidence, then, the true percentage of incident particles reaching the sample was between 24.7% and 26.6%.

* * *

The writers are indebted to Drs. G. Goertzel and Mathew M. Shapiro for numerous helpful suggestions.

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A Nucleonics Survey: Digital Computer Use in the Nuclear Field

Where Digital Computers Are Used in the Nuclear Field

User organization	*Computers owned or rented*	*Outside machines used*	*Future acquisition plans*
ACF Industries	650		650 or 704
Aerojet-General Nucleonics	704		
Alco Products	650		650
Allis Chalmers	650		704
American Machine & Foundry	Datatron 205		
American-Standard		650, 704, 1103A	Datatron
Babcock & Wilcox	Datatron		Bigger
Combustion Engineering	2 650's	UNIVAC, 704	
Curtiss-Wright	704	704	709
David Taylor Model Basin	UNIVAC	UNIVAC, NORC	LARC
General Atomic	650	704	704
General Electric (ANP-Lockland, O.)	650	704	704
General Electric (KAPL)	650, 704		LARC
General Motors	704		
Hanford Atomic Products Op.	702, 650		709
Kaiser Engineers			650
Livermore Laboratory	2 650's, UNIVAC I, 2 704's		LARC
Lockheed Missile Systems Div.	UNIVAC, 650	704	704, 709 LARC, STRETCH
Los Alamos Scientific Lab.	3 704's Maniac I & II		
Martin	704		
NACA	3 650's, 3 604's, 1103		Extension of 650's
New York U.	UNIVAC I	Rarely	
NDA	Datatron	UNIVAC	
Oak Ridge National Lab.	ORACLE	Yes	709
Phillips Petroleum Co.	650		LGP-30 or ALWAC
Raytheon Mfg.	650	ORACLE	704
Savannah River Lab. (DuPont)	650	UNIVAC, 704	650
Westinghouse Bettis Plant	650, 704		Extension of 704

To DOCUMENT the extent of use of digital computers in nuclear engineering NUCLEONICS surveyed the principal atomic-energy establishments in the United States. The results of the survey, tabulated here, show that over 35 medium and large-scale computers are being used for nuclear calculations. In addition more than 20 more will be acquired in the next few years to satisfy the growing demand for computer time. Foremost among the machines ordered for the future are two that represent the ultimate in the present state of the computer art—STRETCH being built by IBM for Los Alamos and LARC, which Remington Rand is constructing for Livermore Laboratory and which is already being eyed by other nuclear groups. (See p. 59 for more on these new computers.)

FIG. 1. IBM 704 electronic data-processing machine uses magnetic cores for internal high-speed storage backed up by magnetic drums. Typical speed is 40,000 instructions/sec

FIG. 2. Remington Rand Univac computer showing central unit in background, supervisory panel in foreground and control printer at right behind operator

Introduction to Analog Computers

Simplicity and speed of analog machines make them better than digitals for problems with few variables and < 0.01 % accuracy

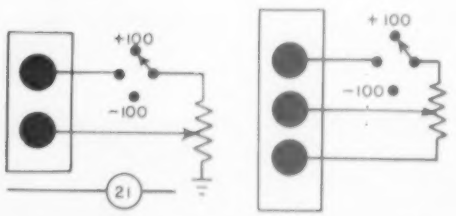

FIG. 1. Potentiometers are used as attenuators and represented by numbered circles

FIG. 2. High-gain d-c amplifier is basic building block of computing circuits

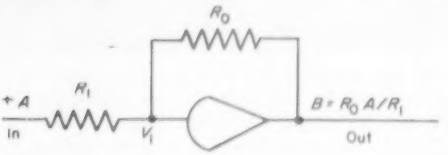

FIG. 3. With feedback, amplifier becomes multiplier (shown) or integrator

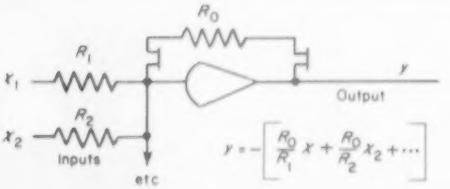

FIG. 4. Parallel inputs permit adding several signals

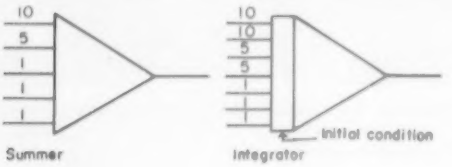

FIG. 5. Symbols for summer and for integrator

AN ANALOG COMPUTER* is a device that uses voltages to represent physical variables. Circuits of the system add, subtract, multiply, and integrate these voltages to solve equations that represent physical problems. Two types of information must be presented to the machine. The first is the equations that describe the problem. Second is the set of numerical values of constants and parameters.

Many people ask, "Which is better, a digital computer or an analog machine?" The choice depends on the problem to be solved. Where part-per-million accuracy is required or where the number of inputs is large (~1,000), a digital computer is necessary. On the other hand an analog machine offers advantages of speed since it performs all of its operations simultaneously rather than in sequence.

Mode of Operation

The basic components of an analog computer are attenuators and operational amplifiers. Attenuators provide the scaling factors that are required, and operational amplifiers of various sorts perform the arithmetic functions of addition, subtraction, etc.

Circuits for attenuators are shown in Fig. 1. A three-position switch allows the operator to connect one end of the potentiometer to one of the machine's standard potentials (typically +100 or −100 volts) or to a point in a patch panel to which a signal voltage can be connected. The output is taken from the center tap. The remaining point is either grounded or left free for other connections depending on the purpose to be served.

The typical operational amplifier, represented as in Fig. 2, is a high-gain d-c amplifier having a flat response (±6%) over a frequency range of 0–20,000 cps. It is usually used in the range 0–100 cps, where its linearity is better than 0.01%. It is typically designed for a gain of 3×10^8 without feedback.

The feedback loop of Fig. 3 makes such an amplifier into a coefficient multiplier, for with feedback the output will be $-R_0/R_1$ times the input.

Since the amplifier draws no current, we have

$$(A - V_1)/R_1 = (V_1 - B)/R_0$$

* This article is based on material supplied by Electronic Associates, Inc., Long Branch, N. J.

Because of the action of the amplifier we have also

$$C = GV_1$$

Solving these equations simultaneously and neglecting $1/G$ in comparison with unity and the ratio of the resistors, we have the result stated above.

If R_0 is made equal to R_1 we have a simple inverter in which the output is the negative of the input. As shown in Fig. 4, we can connect several resistors in parallel with R_1 to make what is called a "summer." The symbol for the summer is shown in Fig. 5. The numbers on the inputs indicate the factors by which each input voltage is multiplied by the action of the circuit.

Integration. If R_0 in Fig. 3 is replaced by a capacitor, the circuit becomes an integrator, and its output is the integral of the input (Fig. 5). The time constant of the circuit can be chosen to suit the problem, or a problem can be rescaled so that existing circuits will handle it. Integrators have "I. C." connections that permit establishment of initial conditions in the solution of differential equations.

Loading is a problem in circuits of this type. An attenuator feeding a high-resistance input (R_1) clearly will respond differently from one that is feeding a low-resistance input. This difficulty is customarily met by feeding the attenuator with a standard signal and adjusting it to give the required output signal.

Servo Multipliers

When multiplication by a variable quantity is required, a circuit such as shown in Fig. 6 is used. Here we require the multiplication of the variable quantities A and B, and, to simplify the circuitry, let us agree to settle for $AB/100$. A is fed into a servo amplifier which compares it with the quantity $100x$ fed back from the first of two potentiometers that are mechanically linked and positioned by the servo motor. The servo motor drives its potentiometers until the signal that is fed back becomes equal to A. When this is accomplished, each of the potentiometers has a setting x such that its output is $A/100$ times its input. Thus the second potentiometer has an output equal to the required $AB/100$. With this circuit all quantities, A, B, and x, can be either positive or negative. Where sign changes are not necessary, the pots need not be double-ended.

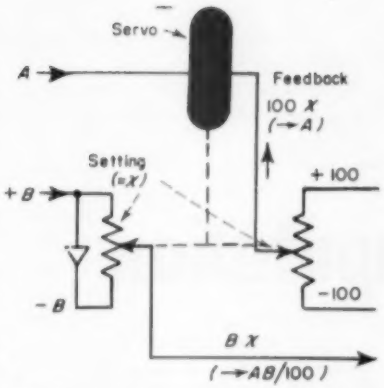

FIG. 6. Multiplication by variable factors is accomplished by servo loop that sets ganged potentiometers in response to feedback

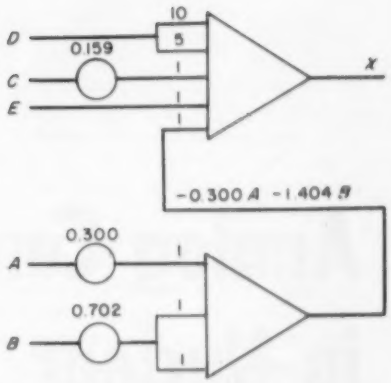

FIG. 7. Circuit consisting of summers and integrators computes value of expression $0.3A + 1.404B - 0.159C - 15D - E$

Typical servo multipliers have one feedback potentiometer and several identically positioned potentiometers mechanically ganged with it. Thus squares and higher powers of a single variable are obtained by feeding one or more of the following potentiometers from preceding ones. As an example, we can obtain $A^2/100$ in the circuit of Fig. 6 by feeding both inputs (A and B) with the voltage representing variable A. If a third potentiometer of the string is fed with $A^2/100$, its output will be $A^3/1{,}000$, etc.

Control

The computer is set up—this is the analogist's term for "programed"—by making connections on a patch panel that is subsequently plugged into the machine. Long and short shielded wires and solid plugs make the required connections.

Operation is governed by a switch having six positions: *standby*, *ref off*, *pot set*, *reset*, *hold*, and *operate*. In *standby* plate supplies are off. In *ref off* plate supplies are on, but the reference supplies (+100 and −100) are off. The switch is in this position when patch cords are removed or inserted in the patch panel. In *pot set* the reference voltages are on, but all amplifier input resistors are disconnected from the inputs and grounded. This is for setting attenuators with proper loading. In *reset* integrators are set to their initial conditions. The problem is started by turning to *operate*. If the operator wants to stop a problem before it is complete, he can do so by turning the switch to *hold*. Then he can measure voltages, locate errors, and determine the magnitudes of different terms of his equations. To continue he switches back to *operate*.

Problem Solution

Combinations of the basic units are used to solve engineering problems. Simple ones require only attenuators and summers. For polynomials and differential equations it is necessary to use servo multipliers and integrators.

Addition-subtraction problems. Problems that involve no more than addition, subtraction, and multiplication by constants can be solved with combinations of attenuators and summers. As an example, let us devise a system to solve the equation

$$x = 0.300A + 1.404B - 0.159C - 15D - E$$

One system that will serve is shown in Fig. 7. The quantities that have negative coefficients go into a summer that inverts them to make them negative in the result. D is fed into the 10 and 5 inputs so that its total contribution is $-15D$ as required by the equation. C is fed to an attenuator that multiplies it by its coefficient, 0.159. A and B are fed to a separate summer, which in turn feeds the first summer. Thus their eventual contribution is positive, which is required. A is multiplied by 0.300 in an attenuator, and B is multiplied by 0.702 and then fed to two unity inputs so that it is eventually multiplied by 1.404.

Differential equations. A simple problem that requires integrators is the falling-body problem. We can write the equation as $d^2x/dt^2 = -g$ and solve it with two integrators and three attenuators. One attenuator selects the value of g that is the input for the first integrator. The other two put in the initial conditions. The first integrator feeds the second, and the output of the second is $x(t)$.

Analog Computation in Nuclear Engineering

Analog computers can solve the kinetic behavior of nuclear reactors in real time, thus enabling the engineer to study that behavior in detail

By S. O. JOHNSON and J. N. GRACE
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FIG. 1. General-purpose dynamic analog computer being used for reactor design at a nuclear engineering center

THE ANALOG COMPUTER is a valuable tool to the nuclear engineer. Reactor transient analysis has been done by analog simulation since the early days of reactor projects. The first simulators for reactors and power plants were special-purpose, "home-made," computers. However, about six years ago general purpose computers as pictured in Fig. 1 began to appear on the market. Today, these "store-bought" computers are in general use in the nuclear engineering field. The purpose of this article is to describe computer uses and limitations in nuclear engineering and to outline simulation techniques, using a pressurized-water reactor plant as an example.

The dynamic analog computer, as distinguished from the passive-network analog, consists of a number of operational amplifiers and multiplication components (*1*). The operational amplifier is a high-gain d-c amplifier that can be used for either integration or summation.* When the output voltage of this amplifier is within the range of linear operation, the input grid voltage is practically zero, because of the high gain. The feedback impedance then determines the transfer function relating output to input. For example, if the feedback impedance is a pure resistance the output voltage is proportional to the sum of all input currents, with phase inversion. With a condenser as the feedback element, the amplifier becomes an integrator. Thus, by the interconnection of summers and integrators, taking care of scaling and phase inversions, a set of ordinary linear differential equations with constant coefficients can be simulated.

Since linear equations with constant coefficients are only of academic interest, multiplication components are needed for the multiplication and division of variables and for raising to powers. The familiar servomultiplier is handicapped somewhat by the speed of response of the servomotor, but has an advantage in that ganged potentiometers allow the multiplication of many variables by another variable that positions the servo. Electronic multipliers are much faster, are more accurate, and, incidentally, are much more expensive. Servomultipliers have been found adequate for the majority of reactor simulation

* For more detail see p. 70 of this issue.

problems; however, there are some multiplications that require the faster response and greater accuracy of the electronic multiplier.

A function generator is needed for the generation of peculiar input functions or for the generation of a nonlinear function of another variable.

The use of a dynamic analog computer is limited at present to the solution of ordinary differential equations. This limitation immediately rules out the solution of neutron diffusion equations for the determination of equilibrium spatial distributions of flux and power within a reactor. These nuclear design problems are generally solved on digital computers, although passive network analogs have been used.

Reactor and plant kinetics equations involving a single independent variable, the time, are ideally suited for analog computation. Ordinary differential equations can be written to approximate the dynamic behavior of nuclear power level, reactor heat transfer, coolant flow and heat transport, pressurizer dynamics, heat exchange, and steam-system kinetics. Some of these processes actually involve distributed parameters and, therefore, partial differential equations; however, difficulty can be avoided by transforming each partial differential equation into a set of ordinary differential equations. For example, distributed heat transfer can be approximated by dividing the heat-transfer element into sections along the direction of coolant flow and then writing ordinary differential equations for the sections.*

Reactor kinetics is a spatial kinetics problem, too. However, the space variations can usually be ignored—the familiar reactor kinetics equations, (*2*), which are ordinary differential equations, have been found adequate to describe reactor transients, with very few exceptions.

Reactor and plant kinetics problems require the solution of a large number of equations simultaneously. Furthermore, nonlinearities and multiplications of variables appear in the equations. These facts preclude the solution by classical means but offer no insurmountable obstacles to analog solution.

Both special-purpose and general-purpose computers have their usefulness. Special-purpose computers are used where little flexibility is required and for training aids, such as portable reactor simulators and the xenon-effect simulator pictured in Fig. 2. Reactor plant kinetics problems generally require the flexibility of a general purpose computer such as that shown in Fig. 1. However, when justified by frequent use, a special-purpose computer, such as a reactor simulator, can be coupled to a general-purpose computer to conserve components.

Suitable Problems

Solutions of the reactor and plant kinetics equations are required in many nuclear engineering problems.

The control engineer must know the self-regulating characteristics (*3*), if any, of the complete system when subjected to various disturbances. He uses a simulator for load transient tests which, together with analytical stability analyses, are used to synthesize an automatic control system (*4*). He must study the response of the system to plant malfunctions, such as the loss of coolant flow or control-rod disturbances, in order to design an emergency shutdown, or scram system (*5*). He also studies reactor startup transients to determine the best procedure and to establish safe startup rates (*6*, *7*).

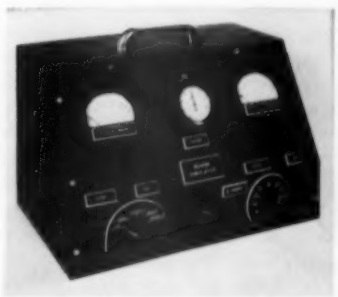

FIG. 2. Portable simulator for effect of fission-product xenon on reactivity

The reactor engineer is concerned with keeping transient temperature variations within established bounds to avoid burnout and thermal-stress problems.

The systems engineer uses a simulator to obtain information of transient changes in coolant volume needed to design a surge pipe and to size a pressurizer.

Reactor kinetics. One of the most frequently encountered reactor-kinetics problems is that of determining the response of the fission power level to arbitrary variations of reactivity. To state the problem mathematically, several ordinary differential equations—the well-known reactor kinetics equations—are required. Because one of the equations contains the product of reactivity and power level, analytic solutions for all but the very simple forms of reactivity variation become very difficult. On an analog computer, integrators, summers, and one multiplier are interconnected to simulate the equations. Input reactivity variations of any form are applied to the multiplier. A voltage analogous to the power level of the reactor then appears at the output of an amplifier and may be recorded or displayed on a meter.

While such a simulation is not a complete representation of a power reactor, in that it does not include heat-transfer and reactivity-feedback effects, it has several useful applications. During startup of power reactors, the power level is, of course, low enough so that no appreciable heating of the moderator occurs, and reactivity feedback effects may be neglected. Hence, simulation of the reactor kinetic equations alone is sufficient to study startup transients over many decades, up to a level where significant power is produced. For "zero-power" reactors, simulation of the reactor kinetics equations alone is sufficient to study the behavior of the reactor under all operating conditions. Hence, the control systems of zero-power reactors can be tested by substituting the simulator for the reactor.

The quantity of analog equipment required to solve the reactor-kinetics equations is not large. If standard general-purpose equipment is used, eight operational amplifiers, one multiplier, and a function generator, which produces a voltage analogous to the reactivity variation, are required. Because the reactor-kinetics simulator is involved in nearly all kinetics studies, it has been found convenient to build a special-purpose simulator, employing two amplifiers rather than eight, to solve this particular set of equations. In an even simpler form, a small portable reactor-kinetics simulator, has been constructed for use as a training aid (*8*). This simulator employs a hand-set potentiometer as a multiplying device. The operator varies the

* An example is provided in the discussion of four-node representation of boiler heat transfer on p. 84 of this report.

reactivity manually and reads the response of the reactor power level on a meter.

With reactivity feedback. In power reactors the problems become more complex. The generation of power by the reactor causes changes in the moderator, resulting in inherent reactivity feedback effects. Hence, a set of equations describing the heat transfer* from fuel to moderator and the transport of heat out of the reactor by the coolant must be coupled to the reactor-kinetics equations. Because each part of a simulator solves a particular equation or set of equations, the coupling within the computer is quite simple, and the analog circuit that simulates the heat transfer equations may be coupled directly to the reactor-kinetics simulator. We then have a simulator that allows the study of a reactor in which the coolant flows on an open-loop basis, e.g., the temperature of the coolant at the reactor outlet has no effect on the coolant temperature at the inlet.

While this does not give a complete picture of present day power reactors, particular types of kinetics problems can be studied without further complication of the simulator. Two examples of such problems are the cold-water accident, in which the reactor-inlet-water temperature is assumed to drop rapidly, affecting the reactivity and causing a rapid increase in power, and the loss-of-flow accident, in which it is assumed that coolant flow decays rapidly, stopping the removal of heat from the reactor.

Plant simulation. For transients lasting longer than one recirculation time of the coolant, it becomes necessary to simulate the heat exchanger and the secondary, or steam system. The equations of heat transfer in the heat exchanger are simulated in a manner similar to that employed for the reactor. This portion of the simulator must be coupled to the reactor-heat-transfer simulator. The coupling must be done through time-delay circuits (*10*). The voltage representing the output temperature of the reactor is delayed and then applied as the voltage representing the inlet temperature of the heat exchanger. Similarly, the voltage analogous to the heat exchanger outlet temperature is delayed and then becomes the inlet temperature of the reactor. These delays represent the time required to transport the coolant through the primary loop piping connecting the reactor to the heat exchanger. Transport delays include an all-pass term and a filtering effect to simulate mixing.

When the heat-exchanger equations are coupled to the reactor-kinetics and heat-transfer equations on an analog computer, the system becomes a power-plant simulator. In addition to the kinetics mentioned previously, the power-plant simulator can be used to study the stability and self-regulating characteristics, the response of the plant to load transients, and all transients resulting from operator or plant malfunctions, such as accidental valve closures, loss of coolant or steam pressure, and accidental rod motion. In addition, the simulation of the basic plant can be complemented by simulation of the pressurizer and the various control systems. It is sometimes used as a substitute for the actual plant in testing and optimizing actual control equipment (Fig. 3). The power plant simulator has also found extensive use in the training of plant operators (*11*).

In its simplest form, the basic power-plant simulator requires a minimum of 50 amplifiers. When greater accuracy is desired, together with the simulation of control systems, etc., the number of operational amplifiers required often exceeds 100.

In addition to the many problems directly related to the kinetic behavior of the plant, several other types of reactor engineering problems may be handled by general or special purpose analog computers. In mobile plants, a variety of shock and vibration problems that effect the operation of the plant are encountered. Many of these problems yield quite readily to simulation techniques. The continuous chemical reactions used to purify the coolant may be studied by simulation. The problem of computing the xenon poison concentration in a reactor after an arbitrary power history is solved quickly by the small portable unit shown in Fig. 3. Analog computers may be applied to advantage whenever a problem involving a set of ordinary differential equations is encountered. They are particularly advantageous when the equations involve nonlinearities.

Advantages and Limitations

Perhaps the chief advantage of an analog computer is its flexibility. Many different problems which can be described by sets of ordinary differential equations may be solved by the same basic group of computing equipment. In present-day computers, all connections between computing components are made on a removable plug board, so that a particular problem can be removed from the computer without destroying the connections representing the problem. Thus, a computer set-up can be conveniently stored for use at some future date.

In addition to being capable of solving a large number of different problems, the analog computer provides a convenient means of studying many facets of the same problem. The response of a particular system to a variety of driving functions may be obtained. Parametric studies are made merely by changing potentiometer settings corresponding to the coefficients of the equations. Thus, the engineer has a simple means of evaluating the effects of design changes on the transient response of a system.

The worth of analog computers is enhanced by their inherent ability to solve problems on a "real" time scale. This property allows their use as testing devices for plant equipment, and incidentally, imparts to the engineer a "feel" for the kinetic behavior of his system.

In other types of problems, the ability of an analog computer to simulate on a time base other than real time often is an advantage. By contracting the time base of the simulator, solutions to inherently slow transients may be obtained without using an undue amount of computer time. For example, in the xenon simulator, the time base is contracted by a factor of 600. Because of accuracy considerations, it is sometimes necessary to expand the time base of the simulator, so that the simulated system operates more slowly than the actual system. Such a procedure is quite often required when the simulator contains servomultipliers.

The maximum voltage variation of

* Because of the necessity of approximating the partial differential equations of heat transfer by a set of differential-difference equations, the amount of analog equipment required to simulate heat transfer in a reactor varies with the type of transient to be studied. A minimum of three operational amplifiers is required, for a one-lump model; but, for very fast transients or transients involving the boiling of the coolant (*9*), as many as 80 amplifiers may be necessary to give the desired accuracy.

analog computers imposes a limitation on the maximum variation of problem variable. In present-day computers, this limitation is approximately 100 to 1. Therefore, some consideration must be given to scaling the problem to the computer. Occasionally this limitation can be circumvented by a transformation of the equations. For example, during the startup of a reactor, the power level may change by many factors of ten. It has been found convenient to transform the reactor kinetics equations so that the variable simulated is the logarithm of the power level rather than the power level itself. In other problems in which there is a wide variation of the variables, it is sometimes necessary to run the problem in several stages. This procedure, of course, requires more computer time.

Accuracy. The accuracy obtainable from an analog computer varies with the size of the problem. While individual component accuracies of the order of one hundredth of a percent are currently obtainable, errors occurring in the computer tend to multiply. Hence, the accuracy deteriorates with increasing problem size. For very large problems a detailed analysis of the methods employed in solving the problem must be made to insure that the greatest accuracy is obtained.

The usual form of the data output of an analog computer offers both advantages and disadvantages. The voltages of primary interest occurring in an analog computer are usually recorded on a strip chart recorder, so that several time dependent functions are automatically plotted in their correct relationships to each other. These recordings are directly analogous to those which would ordinarily be obtained by instrumentation of the physical system. The disadvantage of this type of output is its inherent poor accuracy, and usually is the limiting factor on the accuracy of analog computer solutions. Recently, however, the accuracy of readout has been improved by the development of analog-to-digital converters.

The degree to which an analog computer accurately simulates a system depends of course on the accuracy of the computer in solving the equations and on the accuracy of the equations in representing the system being simulated. While very impressive figures on computer accuracy can be claimed justifiably, the accuracy of the equations, particularly for new types of reactors and plants, is neither as easily defined nor as well understood. It is necessary to emphasize these facts when presenting computer data to compensate for the tendency to accept simulator results as representing perfectly the behavior of the system being simulated.

Analog vs Digital

The question inevitably arises as to whether reactor and plant kinetics equations could not better be solved on a digital computer. To make the correct decision on this question the engineer should try to utilize fully the advantages and disadvantages of each machine. The digital computer can be used to obtain more accurate results, when justified, and can solve any number of similar problems for which a program and code have been developed. On the other hand the analog is more easily programmed and offers greater flexibility with less expenditure of time and effort.

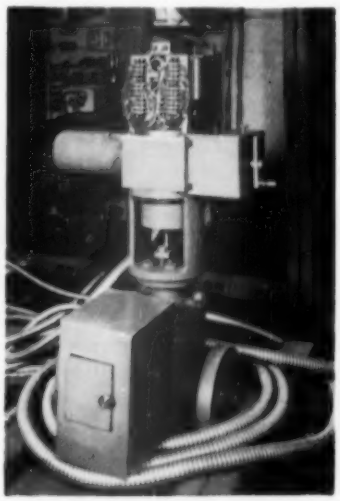

FIG. 3. Control valve coupled to power-plant simulator for testing. Since analog computers can operate in real time they can be used for such component testing

Because of these considerations it has been found that new problems in reactor and plant kinetics are better solved on the analog. The engineer obtains a better feel for the problem as to the adequacy of his model and equations and can take advantage of simplicity and flexibility to study modifications of the system being simulated. On the other hand, when some problems become routine, such that many repeated solutions for slightly different designs are required, the development of a digital code may be justified.

Open vs Closed Shop

As stated earlier an analog computer affords the engineer an opportunity to get a feel for the behavior of the system being simulated in addition to providing quantitative data. Flexibility allows changes in plan when desired to avoid fruitless pursuits and the running of useless data. Therefore, to take full advantage of an analog computer and in the interest of economy it is very desirable that the engineer work in the laboratory when his problems are being simulated. This is obviously necessary when a new type of reactor problem is being analyzed or when new simulation techniques are involved. Routine problems can be set up and run by trained technicians; however, it is desirable that the responsible engineer be present if computer time and data processing are to be conserved. The alternative procedure of maintaining a completely closed shop, involving problem requests on a requisition basis, can result in a large number of problem requests to cover broadly all areas of possible interest. This "shotgun" approach is very uneconomical of computer time and data processing.

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Dynamic Simulation of a Fast Reactor

Two hypothetical reactor accidents, startup and loss-of-pump power, are analyzed with an analog computer

By R. G. OLSON
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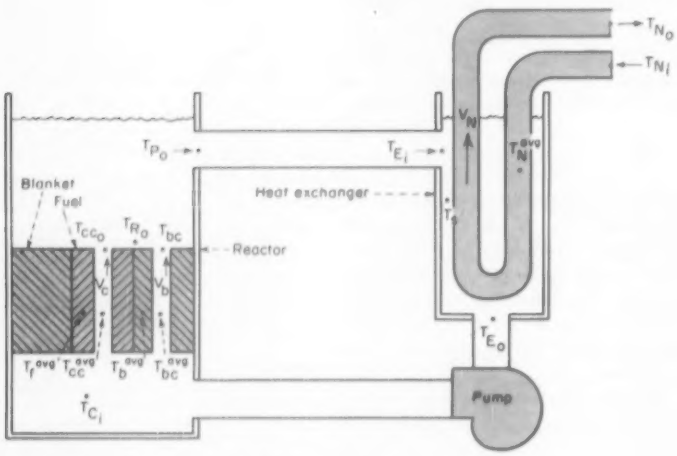


FIG. 1. Model for simulation of fast-reactor system

SAFETY is an important consideration in the design of a nuclear reactor. To ensure safety, hypothetical accidents must be studied and analyzed. Analog simulators afford an excellent means of studying such accidents without actually having to produce them.

While the analysis of accidents is important in the design of any reactor system, it is especially so for fast reactors because even small geometry changes due to temperature effects produce significant reactivity changes.

The Atomic Power Development Associates' simulation program, started in 1954, has shown that the Enrico Fermi Fast Breeder, as designed, is capable of accommodating any of the assumed accidents (*1*). The two extreme accidents, the startup accident and loss-of-pump-power accident,* studied in this article, however, did admit of possible melting in the center of some fuel pins.

Over-All Considerations

Though the dynamic response of fast reactors in the delayed-critical range is similar to that of thermal reactors, fast reactors have the following four inherent dynamic responses that are different:

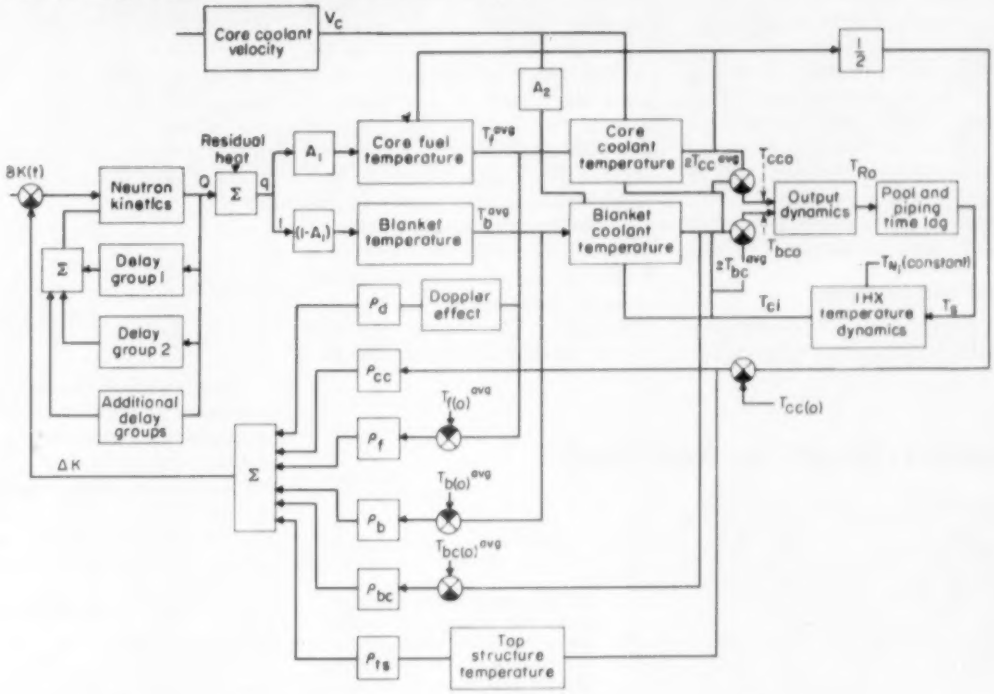
1. The lifetime of prompt neutrons in fast reactors is $\sim 10^{-7}$ sec. This is several orders of magnitude smaller than in thermal reactors. The prompt lifetime is important only when prompt critical is approached or exceeded but is of little importance otherwise. The other time constants, such as delayed-neutron emission, heat conduction, mechanical motion, and heat transport, are several orders of magnitude larger than the prompt lifetime.

2. The temperature coefficients of reactivity for fast reactors are an order of magnitude smaller than those for thermal reactors. No fast-acting, dominant temperature coefficient overrides the others, such as moderator expansion in a thermal reactor. If some of the coefficients are positive or have large characteristic time lags, the reactor may possibly display instability (*2*).

3. The delayed-neutron fraction β for fast fission of U^{235} is about 8% smaller than for thermal fission of U^{235} (*3*). Although β would be smaller in a fully enriched reactor, the presence of U^{238}, which has a β of 0.0176, in a less

* They are extreme in that they require the simultaneous malfunction of the operator and the control equipment to occur.

This dynamic simulator . . .



. . . solves these reactor-kinetic equations . . .

The time-dependent equations of reactor power are

$$\frac{dQ}{dt} = \frac{\delta k - \beta}{l} Q + \sum_{i=1}^{6} \lambda_i P_i + \frac{\Delta K}{l} Q$$

$$\frac{dP_i}{dt} = \frac{\beta_i}{l} Q - \lambda_i P_i$$

where

Q = reactor power
δk = excess multiplication factor
β = delayed-neutron fraction
λ_i = decay constant of ith group of delayed neutron emitters
P_i = contribution to reactor power from fissions caused by ith group of delayed neutrons
l = prompt-neutron lifetime

For the Enrico Fermi fast reactor $l = 1.8 \times 10^{-7}$ sec.

The driving function is a 1 cent/sec ramp in the startup accident

$$\delta k = R(t) = 0.000075\, t$$

or, for flow decay,

$$v = f(t)$$

where $f(t)$ is the slope of the pump slowdown characteristic (5).

The temperature effect on reactivity is given by

$$\frac{\Delta K}{K} = \sum_n \rho_n [T_n^{avg} - T_n(O)]$$

Temperature coefficients to be summed over are*

$$\rho_f = -1.72 \times 10^{-6}/^\circ \text{F}$$
$$\rho_b = -0.444 \times 10^{-6}$$
$$\rho_{ts} = -1.777 \times 10^{-6}$$
$$\rho_{cc} = -3.94 \times 10^{-6}$$
$$\rho_{bc} = -1.83 \times 10^{-6}$$

For the core, the thermal equations connecting the mean fuel temperature with the power are

$$M_f C_f \frac{dT_f^{avg}}{dt} = q_c - A_{f-c} U_{f-c} (T_f^{avg} - T_{cc}^{avg})$$

and for the blanket

$$M_b C_b \frac{dT_b^{avg}}{dt} = q_b - A_{b-c} U_{b-c} (T_b^{avg} - T_{bc}^{avg})$$

The equations of heat transfer from fuel to core coolant are

$$M_{cc} C_c \frac{dT_{cc}^{avg}}{dt} = A_{f-c} U_{f-c} (T_f^{avg} - T_{cc}^{avg}) - 2a_c \delta_c v_c C_c (T_{cc}^{avg} - T_{ci})$$

and in the blanket

$$M_{bc} C_c \frac{dT_{bc}^{avg}}{dt} = A_{b-c} U_{b-c} (T_b - T_{bc}^{avg}) - 2a_b \delta_c v_b (T_{bc} - T_{ci})$$

The mixing of coolant out of the core and blanket is given by

$$T_{R_c}(t) = \frac{a_c}{a_c + k_m a_{bt}} T_{cc_o}(t) + \frac{k_m a_{bt}}{a_c + k_m a_{bt}} T_{bc_o}(t)$$

The temperature input to the heat-exchanger input sodium is

$$\tau \frac{dT_{E_i}}{dt} = [T_{R_c} - T_{R_c}(O)] - [T_{E_i} - T_{E_i}(O)]$$

The heat transfer from the heat exchanger is

$$M_e C_e \frac{dT_e}{dt} = g(T_{E_i} - T_S^{avg}) - h(T_{E_i} - T_{Ni})$$

where

$$T_e = \frac{T_{E_i} + T_{E_o}}{2}$$

* For symbols not explained on this page, see text, pp. 78 and 79.

Reactor-Startup Accident

than fully enriched reactor will tend to increase the effective delayed-neutron fraction.

4. The total excess reactivity necessary for a fast reactor is of the order of one dollar. This is small because (a) fission-product poisoning is very small owing to small absorption cross sections of fission products for fast neutrons, and (b) temperature override is small because of small temperature effects.

System and Its Model

The reactor system is shown schematically in Fig. 1. The electronic analog is shown in block form on p. 77. The three primary coolant loops are simulated as one. Liquid sodium enters the bottom of the reactor vessel with a temperature T_{c_i} and passes up through the heat-transfer channels. The coolant velocity through the core is v_c, and the coolant velocity through the blanket is v_b. During the passage of sodium through the core and blanket, it is heated by nuclear heat generation in the fuel and blanket material and leaves at temperatures $T_{c_{c_o}}$ and $T_{b_{c_o}}$, respectively. The sodium then goes into an upper reactor pool that contains ~13,700 gal of primary coolant.

A lumped-point heat-transfer model is used. A linear temperature variation between inlet and outlet coolant temperatures is assumed in both the core and blanket elements, so that the average coolant temperature of the core $T_{c_c}^{\text{avg}}$ and of the blanket $T_{b_c}^{\text{avg}}$ is given by

$$T_{c_c}^{\text{avg}} = \frac{T_{c_{c_o}} + T_{c_i}}{2}$$

and

$$T_{b_c}^{\text{avg}} = \frac{T_{b_{c_o}} + T_{c_i}}{2}$$

The average fuel temperature T_f^{avg} is determined by the average core coolant temperature and the rate of generation of fission heat in the core. For a 300-Mw reactor, rated-load heat generation in the core q_c (rated) is 275 Mw. The 91 core subassemblies are simulated as one subassembly with average heat generation.

The average blanket temperature T_b^{avg} is determined by the average blanket coolant temperature and the rate of generation of fission heat in the blanket. Rated-load heat generation in the blanket is based on an average blanket power q_b (rated) of 25 Mw. The 572 blanket subassemblies are simulated as one subassembly that has the proper reactivity effect on the reactor but is not a blanket subassembly with average heat.

The outlet coolant temperature of the reactor T_{R_o} is determined by the core and blanket outlet coolant temperatures weighted by their coolant mass flow rates.

The reactor pool time lag and piping time delays are approximated by a lumped first-order time lag τ that varies with the coolant velocity. The coolant inlet temperature to the reactor T_{c_i} was assumed identical, for this study, to the coolant outlet temperature of the intermediate heat exchanger.

For simplicity in representing heat transfer to the secondary loop, it is assumed that the primary coolant temperature at the inlet of the intermediate heat exchanger T_{E_i} is equal to the secondary coolant outlet temperature T_{N_o}.

Since the total primary coolant loop transport time delay has been approximated by a lumped first-order time lag on the primary coolant outlet temperature, the primary coolant temperature at the outlet of the heat exchanger T_{E_o} is the same as the primary coolant temperature at the reactor inlet T_{c_i}.

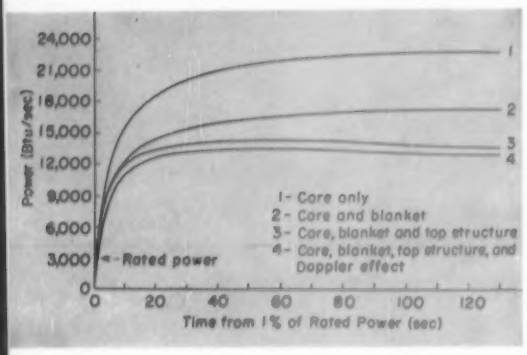


FIG. 2. Maximum power drops from 7.5 times rated to 4.5 times rated as extra effects are considered. Note that the blanket and top structure temperature coefficients have considerable effect. Doppler effect, however, changes the rising front but has little effect on final power because temperature rise decreases its effectiveness

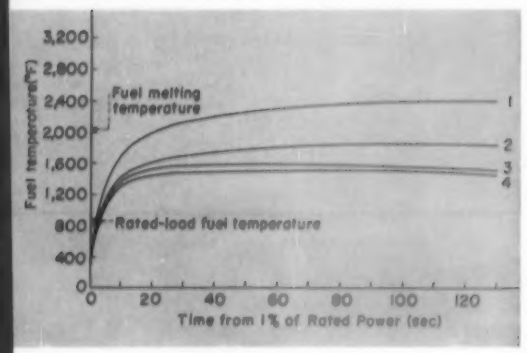


FIG. 3. Average fuel temperature increases at first, then decreases due to the temperature coefficients during startup accident. As would be expected, blanket and top structure are the dominant effects, reducing the average fuel temperature well below the fuel melting temperature of 2,070° F

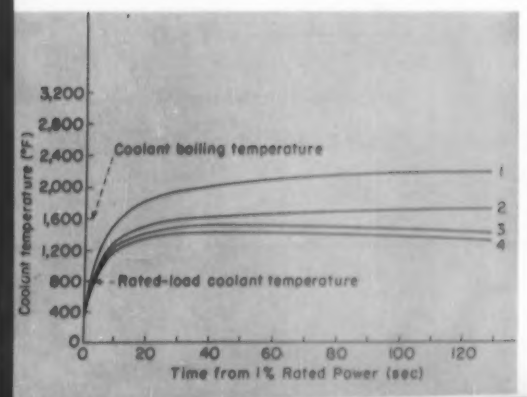


FIG. 4. Temperature of reactor coolant leaving core for startup accident. Added temperature effects reduce average coolant temperature 200° F below boiling point of sodium (1,630° F)

Loss-of-Pump-Power Accident

Heat is assumed to be taken out of the primary system by the heat exchanger at the same rate as it enters in the core. Since the transit time of the coolant around the primary loop is about 60 sec at rated flow, the reactor coolant inlet temperature T_{ci} can be considered constant for the first 60 sec.

As diagrammed on p. 77, the fission heat generated in the core and blanket are dependent on the externally variable excess multiplication factor δk; on the temperature coefficients of reactivity of the fuel ρ_f, of the core coolant ρ_{cc}, of the blanket material ρ_b, of the blanket coolant ρ_{bc}, of the top structure ρ_{ts}; and on their respective temperatures.

Gamma heating was assumed to contribute at a constant rate to the total heat production upon shutdown; specifically it is taken to be 4% of the heat production of the reactor at rated load.

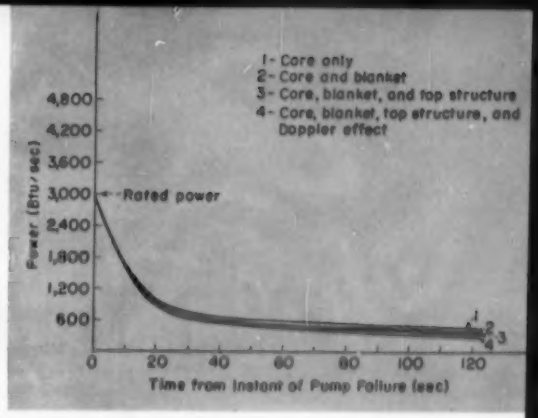


FIG. 5. Power drops as flow decays with inertia decay of pump motor. Note that temperature coefficients have small effect on power decay. Coolant flow is most important factor

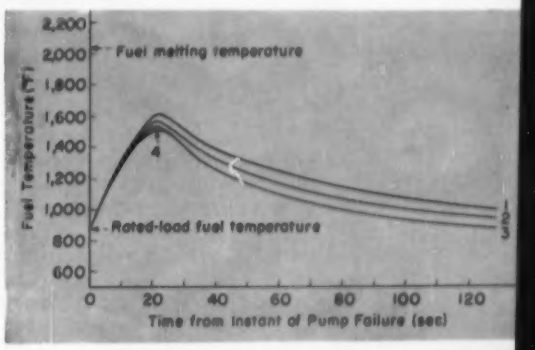


FIG. 6. Average fuel temperature decreases 100° F on peak due to adding blanket, top structure and Doppler effects. Fuel temperature is well below melting point, 2,070° F

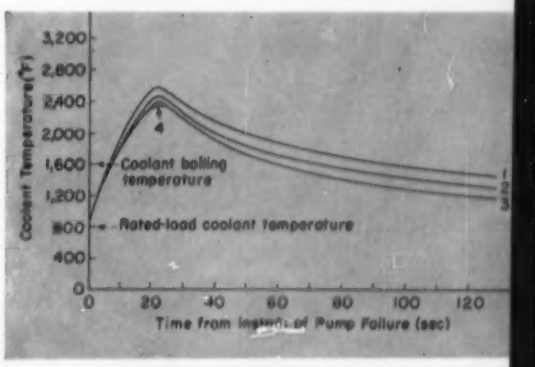


FIG. 7. Outlet coolant temperature rises fast because of velocity decay. Value of peak depends on thermal circulation value of coolant

Mathematical Formulation

The schematic and equations describing the system shown in Fig. 1 are given in the box on p. 77.

The 17 simultaneous equations are solved for a linear input of reactivity or coolant flow decay with four different combinations of temperature coefficients: (a) core only; (b) core and blanket; (c) core, blanket, and top structure; and (d) core, blanket, top structure, and Doppler effect. The results are shown in Figs. 2–7 with the four curves in each figure, from top to bottom, reflecting the cumulative insertion of the temperature coefficients.

Conclusion

This study indicates that, as with all reactors, the control system must be relied on to prevent burnout and, therefore, should be made to be fail-safe. It also indicates that in increasing power accidents the extra temperature coefficients contribute substantially to limiting the power excursion, but that in decreasing power accidents they have small effect on the results.

In addition, considerable study of temperature coefficient effects is necessary in fast-reactor design. Although the temperature coefficients are smaller than for thermal reactors, the allowable coolant temperatures are higher. If the coolant curves for fast and thermal reactors are normalized to their coolant boiling temperatures, the coolant performances are similar.

Primary coolant boiling was not represented in the simulation, therefore, these results do not hold after the sodium begins to boil. However, coolant boiling does produce negative reactivity effects in the Enrico Fermi reactor, hence, the results are conservative.

It must be remembered that the values used for temperature coefficients are uncertain. We believe they are conservative except for that of the Doppler effect. The Doppler effect was added to give a complete picture, even though its magnitude is uncertain.

* * *

This work was done while on leave of absence from the Detroit Edison Co.

The author is indebted to L. L. Kintner for providing the heat transfer calculations, and to R. B. Nicholson for the determination of a blanket subassembly which provided realistic blanket temperature reactivity effects. The analog computer setup and operation were performed by Earl Lemon of the Bendix Aviation Corp.

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PWR Training Simulator

Commercial analog computing units simulate PWR nuclear and thermal characteristics in this unit, which will train operators in normal as well as abnormal operation

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FIG. 1. Console of training simulator, similar to actual PWR console, has plant loops and main components laid out schematically on operating panel with controls at appropriate positions on diagram. Components and controls not fully simulated are included in dummy form. Meters display temperatures, flows, pressures, and nuclear aspects. Annunciator indicators above meters operate when alarm condition arises. Computing equipment is housed separately in cabinets (right). Sixty operational amplifiers, four servo multipliers, voltage reference supply, and power supplies are commercial analog-computer units costing about $25,000. Passive computing elements are mounted in these racks. Relays and motor-driven potentiometers are mounted under console to minimize interconnecting wiring

THE PWR TRAINING SIMULATOR is a tool to train personnel who are to operate the Shippingport Atomic Power Station.* With a control console (see Fig. 1) similar to that of the power station, it provides an impact not obtainable from any amount of verbal instruction.

The pressurized-water reactor is inherently stable—the operator has time to check before he throws a switch or turns a knob. This contributes to achieving realism with the simulator at moderate cost.

The emphasis in designing the simulator has been placed on reactor operation; other parts of the main console have been modified. Thus the training-console length is less than that of the actual plant console.

Control and indication of the electrical and steam systems (see Fig. 2 for schematic of PWR plant coolant and steam cycles) have been simplified or eliminated if their effect on reactor operation is relatively small or represents a functional duplication. Thus coolant-flow simulation is restricted to controllable flow in one loop and fixed flow in the remaining three loops; one circuit simulates the fixed-flow loops.

Performance

The PWR training simulator derives a considerable amount of realism from direct and continuous solution of the equations of the simulated plant. This realism is valuable when the subsequent plant operation depends on what action is taken by the operator or when action taken by the operator depends on what the plant appears to be doing. A typical simulated plant operation is shown in Fig. 3.

A remote-control panel located at the computer racks enables the instructor to inject various "trouble" conditions by tripping a primary pump, safety shutdown (scram), boiler feed pumps, or turbine throttle. He may also inject reactivity of various rates of change to simulate xenon transient conditions. In addition, trainee performance can be checked by plugging recorders into test sockets on the computer.

Simulator Capability

Simulation of the following plant op-

* J. W. Simpson, Milton Shaw. PWR power plant, part 1—nuclear power generator, *Westinghouse Engineer*, **15**, No. 6, 178 (1955).

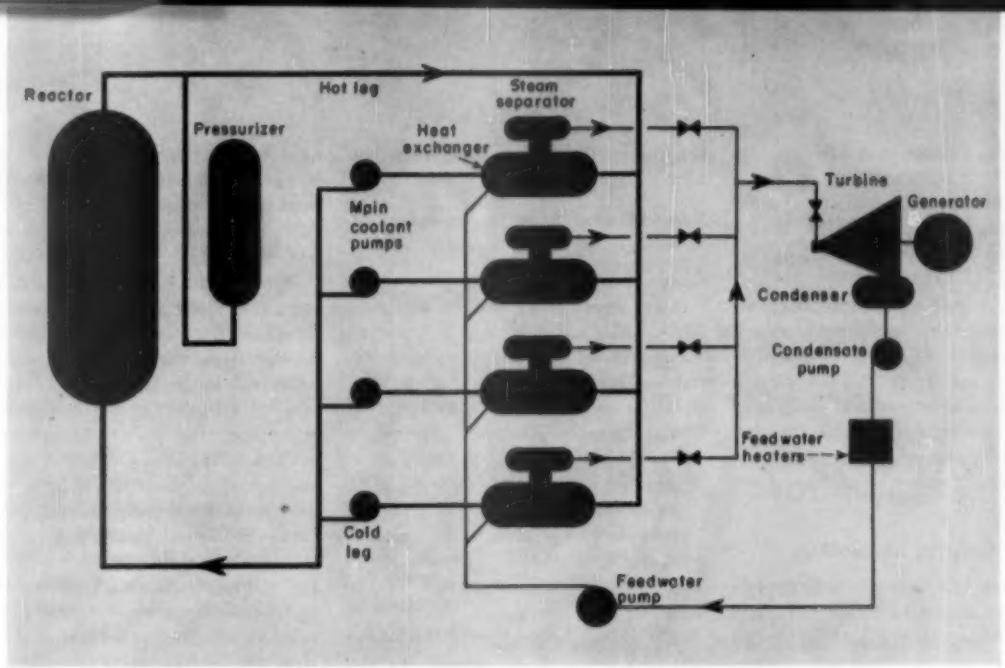

FIG. 2. Flow diagram of PWR reactor and steam plant

erations are provided by the training device:

Reactor nuclear startup.
Control-rod adjustment.
Plant warmup and cooldown.
Control of pressurizer pressure.
Control of pressurizer level.
Shutdown or startup of coolant loop.
Emergency turbine shutdown.
Emergency reactor shutdown.
Turbine-governor control.
Drum-level and feedwater-valve control.
Decay-heat valve control.
Steam-stop valve control.
Shutdown or startup of boiler feed.
Linear-reactivity disturbances (xenon effects).

Reactor Simulation

Reactor simulators are usually restricted to a neutron-flux range of about three decades—between the noise level and the saturation level of the computing operational amplifiers. In most engineering simulation studies, the range of reactor power to be studied can be covered by rescaling to stay within the operating range of the amplifiers, and so this restriction is of little consequence.

In the training simulator, however, it was desired to provide a wide range of reactor-power-level simulation to include both nuclear startup and operating-power level with continuity between ranges. This was accomplished by using two reactor simulators, one for startup range and one for operating range. These are interlocked at a fixed ratio by an operational amplifier and voltage divider.

In effect, while in the startup range, the startup simulator is in control and provides output to the nuclear-instrument indication, while at the same time it sets the initial conditions for the operating-range simulator. After an automatic relay circuit switches control over to the operating-range simulator at the proper point, the latter simulator sets the initial conditions on the startup simulator and applies a nuclear-level signal to the plant computer. A limiter clips the startup-simulator output above the switchover point but below the startup-range amplifiers' saturation level.

By this scheme, simulation of increasing reactor power is exact. On decreasing power, the simulation is not exact as a result of the startup-range amplifiers having been held at constant voltage by the limiter while the operating-range output was decreasing. This error is small for slow power changes, that is, when the time between the point at which the startup level falls below the limiter threshold and the point at which the simulator is switched to startup-range control is long compared with the longest time constant (~80 sec) in the reactor-simulator circuit. A switching arrangement to avoid this situation is possible, but for this application it was not considered worth the additional circuit complication.

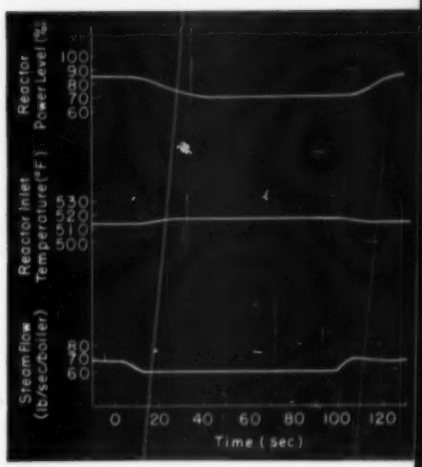

FIG. 3. Simulated plant reaction to 12% decrease in steam flow from full-load conditions

The scaling factor used between the two reactor simulators was arbitrarily set at 100:1 (that is, the startup-range neutron-level signal is 100 times the operating-range neutron-level signal up to the limiter threshold). A considerable range of ratios is permissible, but 100:1 was selected in keeping with the desired accuracy and the noise level of the amplifiers.

The reactor simulator gives good results over a range of approximately

six decades. This coverage can be extended by cascading additional simulators, but six-decade simulation, while not covering the full range of the actual plant, was considered sufficient for training.

Temperature- and pressure-coefficient effects on the reactor are provided in both ranges by a servo-multiplier-controlled reactivity feedback. No xenon computer is used, but positive or negative reactivity can be inserted manually by the instructor from a remote-control panel to simulate xenon effects.

Reactor-Control Simulation

An automatic rod-control system in the reactor simulator provides control by both temperature and rate of change of neutron-flux level, in the same manner as provided for in the actual PWR power plant.

Signals representing the reactor inlet and outlet temperatures are averaged in a summing amplifier to obtain average temperature. A constant signal representing the reference temperature of 525° F is subtracted from the average temperature, so that positive and negative excursions from 525° F are represented by positive and negative output voltages.

An additional signal of the form $(K/n)dn/dt$ (n representing neutron-flux level) is added to the temperature-control signal. This signal is obtained by differentiating and then filtering the output of a logarithmic amplifier fed directly from the linear neutron-level signal.

The resulting control signal fed to the rod-control relays then takes the form

$$\text{Error signal (° F)} = T_{avg} - T_{ref} + (K/n)dn/dt$$

where the constant K relates the derivative signal to Fahrenheit temperatures. The rod-control relays are set to actuate rod insertion at an error-signal voltage representing a 3° F temperature increase, and withdrawal at 3° F temperature decrease. The relays release when corrective rod motion brings the error signal back to ±2° F.

As designed, the external temperature-control loop permits the negative temperature coefficient of reactivity to adjust power without motion of the rods for load changes of the order of 10–20% of full-power rating. For larger load changes, the external loop assists the temperature coefficient by introducing rod motion to make a more rapid adjustment of reactor power.

A three-position manual rod control (rods in, hold, or out) is available at the console. The operator controls reactivity input to the reactor simulators by energizing a motor to drive the reactivity-input potentiometers.

Automatic safety insertion of rods at normal rates is provided for excessive power-to-flow ratio. The signal overrides both automatic and manual rod control to provide safety backup equivalent to that of the actual plant. Safety shutdown is controlled by a high-speed drive motor that drives the reactivity potentiometers to the full negative, shutdown position at speeds 100 or more times greater than normal rod speed. Safety shutdown is actuated by excessive power-to-flow ratio, low primary-system pressure, and high reactor-outlet temperature.

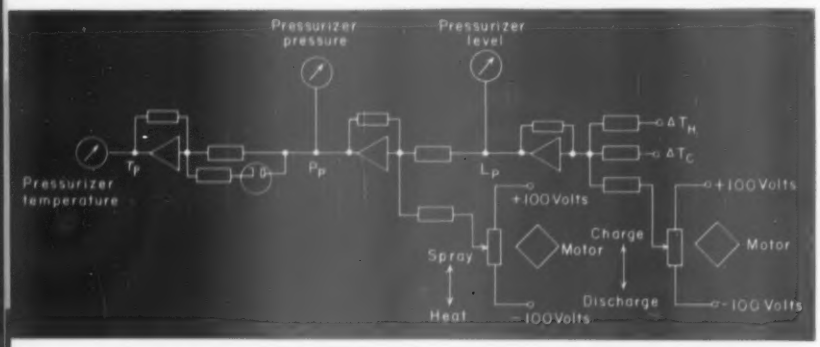

FIG. 4. Pressurizer simulation system: $L_P = 1.75\,\Delta T_H + 1.25\,\Delta T_C$ + charge − discharge; $P_P = 14.7 L_P$ + heat − spray; T_P = nonlinear function × pressurizer pressure; $\Delta T_H = T_H - 475°$ F; $\Delta T_C = 475°$ F $- T_C$

Primary-Loop Simulation

The heat-transfer characteristics of the reactor are duplicated by an analog computer that continuously solves the reactor heat-transfer equations. It provides coolant-temperature information for display and for the boiler simulators.

Coolant-transport delay times are simulated by electronic delays, which are grouped into two circuits, one between the reactor outlet and boiler inlets, the other between the boiler outlets and reactor inlet. It is possible to restrict the frequency response of these delays considerably, in view of the fact that the information is presented only in the form of visual meter readings, causing small variations, especially variations with respect to time only, to be unnoticeable. The delays in the training simulator give accurate transient response only for angular frequencies less than 1 cycle/sec, resulting in a considerable reduction in the number of amplifiers required, while not noticeably deteriorating the accuracy of the information presented.

Pressurizer

Changes in the average core temperature due to power excursions and changes in reactivity, and subsequent correction by control-rod movement make the system coolant volume a variable. These volume changes would cause wide variations in coolant-system pressure if the plant were operated as a solid system.

The function of the pressurizer is to regulate the system pressure within a lower limit set by the reactor hot-spot temperature, and an upper limit determined by the safety- and relief-valve settings. The pressurizing system maintains a 2,000-psi saturation steam heat in a separately heated pressurizing vessel. Pressure can be increased by heat from electric immersion heaters. During a surge, pressure can be reduced by spraying cooler water through nozzles at the top of the vessel.

Pressurizer level is simulated (Fig. 4) as a function of primary coolant temperature and the contribution made by a motor-driven potentiometer that simulates charging and discharging of the primary system. For simplicity, the charge and discharge rates were

assumed equal. Thus, with a single-speed reversible motor, charging is given by one direction of potentiometer travel and discharging by the other direction.

The effects on pressurizer pressure of level and heater-spray action are provided. The heater-spray system is a motor-driven potentiometer similar to the one used in the charge-discharge system. Use of a system of this type implies that there is no heat loss in the pressurizer or that heat loss is balanced by backup heaters.

The steam-table relationship between pressurizer pressure and temperature is simulated approximately by a diode nonlinear-function generator, in which the curve is approximated by two straight-line segments.

Boiler Simulation

Solution of the boiler heat-transfer equations is accomplished in two circuits—one for the variable-flow loop and one for the three fixed loops. The change in heat-transfer conditions between full flow and no flow in the variable-flow loop is accomplished by relay switching. Telephone-type relays on plug-in assemblies provide reliable switching in these circuits.

Steam-load simulation is provided to act as a turbine load, although no turbine characteristics are simulated. In this case, steam load is assumed to be proportional to steam pressure and throttle setting. Governor characteristics are simulated to the extent that when the steam stop valve is closed in the variable loop, the loss of steam is made up from the other three boilers. Generator output is assumed to be proportional to steam flow, above a minimum level of steam flow. Below this level, generator output is zero. In addition to turbine steam load, a decay heat valve provides a small steam load (about 2% of rated output) when open above 707-psi line steam pressure.

Boiler Feed System

The boiler in the variable loop has boiler-feed simulation. This system integrates the difference between boiler steam flow and feedwater flow to obtain drum level.

An additional input to the drum-level amplifier is used for the boiler swell characteristic. The automatic boiler-level control, a three-element control, uses a servo multiplier to simulate the automatic feedwater valve system. Two-pump boiler feed is simulated, including pump-loading characteristics.

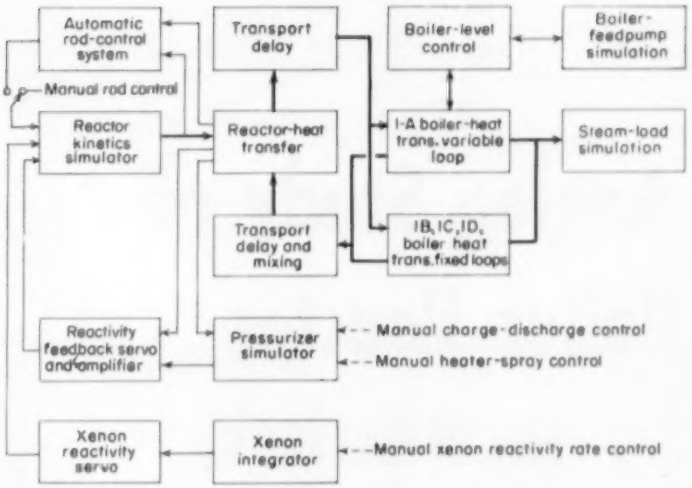

FIG. 5. Diagram of training simulator; plug-in units simplify maintenance

Alarm and Annunciator

A thyratron-controlled annunciator unit, located at the console, provides ten annunciator windows and an audible alarm for the following abnormal conditions:

(1) 1A loop low coolant flow, (2) safety insertion, (3) safety shutdown, (4) high reactor-outlet temperature, (5) high or low pressurizer pressure, (6) high or low pressurizer level, (7) low-pressure protection system disabled (manually), (8) throttle trip, (9) high or low boiler-drum level, (10) boiler feedpump breaker trip.

In addition, high-neutron-flux-level trips provide a safety insertion at normal rod rate and a safety shutdown. Safety shutdown can also be tripped by high reactor outlet temperature and by low pressurizer pressure.

Construction Features

The simulator design (Fig. 5) provides for ease of maintenance, essential with commercial electronic equipment, and simplification wherever possible.

All essential components, including relays and nonlinear circuit units, are plug-in. This simplifies replacement for maintenance and changes in plant parameters, and allows isolation of a defective circuit for trouble shooting by unplugging its input and output components. The sockets for the plug-in components serve as test points where circuit voltages can be read by plugging in a meter lead. This arrangement allows an organized system of maintenance checks to be set up by specifying numbered test points and corresponding correct voltages or waveforms.

In several cases where simulator outputs are to be added, with the result to be used for meter display only, it was found convenient to add currents at the meter, thereby eliminating a summing amplifier. An example is the total-feedwater-flow meter, in which boiler 1A feedwater flow and boilers 1B, 1C, and 1D steam flow (assumed equal to the feedwater flow for these boilers) are added.

Insofar as possible, the solution of nonlinear equations is avoided in the training simulator to minimize the circuit complexities involved with nonlinear circuit elements. Diode-type nonlinear function generators, when used, are assembled in standard interlocking metal boxes with plug-in bases and provide ease of maintenance and adjustment.

In the training simulator, it was possible to achieve satisfactory results from nonlinear functions approximated by a minimum number of straight-line segments, usually three or four. In more accurate engineering studies, a considerably larger number of straight-line segments would be required.

* * *

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Analog Study of Boiler-Reactor Interaction

The desirability of four-node over one-node boiler representation is brought out in this study of the interaction between boiler and reactor

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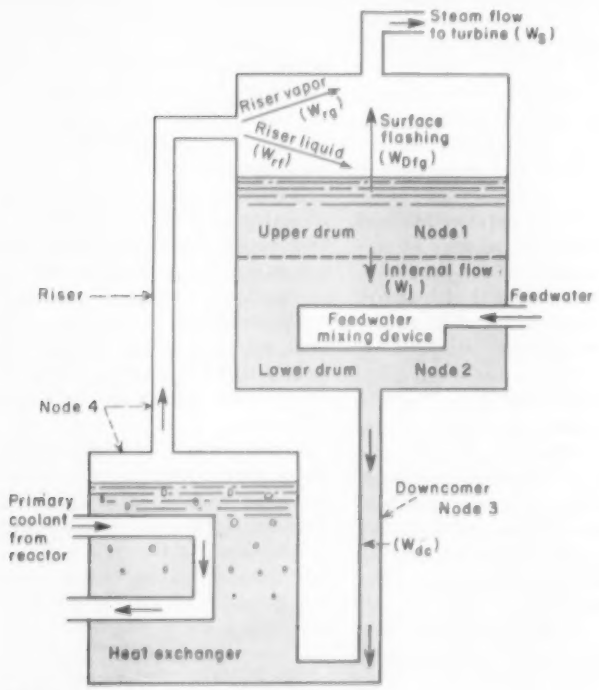


FIG. 1. Boiler is represented by four nodes

IN NUCLEAR POWER PLANTS where the reactor temperature coefficient of reactivity is moderately negative, the reactor can be controlled in part by the natural response of reactivity to temperature changes. The temperature changes and, hence, reactor power level disturbances originate in the steam side of the plant. Boiler operation affects reactor operation.

A standard, general-purpose analog computer made by Electronic Associates, Inc. was used to study interaction between boiler and reactor. The entire study required 72 amplifiers, of which 32 were integrators. Seven servo-multipliers and 4 function generators were required to represent the nonlinearities. The 138 scale-factor potentiometers used could have been reduced in number by using fixed-purpose circuits, but potentiometers preserve the flexibility of the equipment for other studies.

Boiler Model

The boiler consists of heat exchanger, riser, steam drum, feedwater mixing device and downcomer. The general arrangement of these components is shown in Fig. 1. Subcooled water enters the heat exchanger through the downcomer and is heated to boiling. The output that goes up the riser is a mixture of steam and water. At the top of the drum, the steam is separated from the water and proceeds to the turbine while the water falls into the drum, where it mixes with incoming feedwater and returns by natural convection to the heat exchanger. Natural convection is maintained because the subcooled downcomer is more dense than the steam-water mixture in the riser.

Representation by nodes. The continuous or distributed heat transfer phenomena of the boiler involve partial differential equations, the exact solution of which are not obtained on analog, or digital, computers. Ordinary differential equations can be written and solved for these phenomena, however, by assuming lumped-parameter nodes. The nodes represent interconnected energy-storage elements and, hence, simulate the original continuous heat-transfer phenomena. The simplest method of representing the boiler is to consider it as one node, well

* Operated by General Electric Co. for AEC.

mixed and at a uniform temperature throughout.

Four-node boiler. A more sophisticated four-node model can be used. Mass and energy balances are made on each node in the boiler, and the analysis is completed by obtaining the fluid flow between the nodes. The nonlinearities in the two-phase riser flow equations are not easily adaptable to common analog computing techniques. Therefore, the exact analysis is approximated by one in which the downcomer flow and heat exchanger mass are arbitrary inputs obtained from steady-state analyses.

Energy Balances

In the four-node model (Fig. 1), node 1 represents the energy balance for the upper portion of the steam drum, node 2 is for the lower portion of the steam drum, node 3 the downcomer, and node 4 the heat exchanger and riser.

Steam drum. The majority of the steam drum's heat capacity is in the liquid at the bottom of the drum. Injection of the feedwater into this particular boiler subcools only the lower portion of the steam drum and the downcomer. Thus the steam drum contains saturated liquid in addition to varying degrees of subcooled liquid.

This analysis arbitrarily divides the drum into a saturated region of variable volume and a subcooled region of fixed volume with perfect mixing occurring within each node. The two steam drum energy balances are

$$d(\gamma M_{D1}\Theta_{D1})/dt = w_{rf}h_{rf} - w_i h_{D1} - w_{Dfg}h_g \quad (1)$$

and

$$\gamma M_{D2}(d\Theta_{D2}/dt) = w_i h_{D1} + w_{fw}h_{fw} - w_{dc}h_{D2} \quad (2)$$

where

γ = specific heat (Btu/lb-° F)
M_{D1} = upper drum (node 1) liquid mass (lb)
Θ_{D1} = upper drum (node 1) liquid temperature (° F)
w_{rf} = riser liquid flow rate (lb/sec)
h_{rf} = riser liquid enthalpy (Btu/lb)
w_i = internal steam-drum flow rate
h_{D1} = upper drum (node 1) liquid enthalpy
w_{Dfg} = upper drum (node 1) flashing rate (lb/sec)
M_{D2} = lower drum (node 2) liquid mass
Θ_{D2} = lower drum (node 2) liquid temperature
h_g = enthalpy of saturated steam
w_{fw} = feedwater flow rate
h_{fw} = enthalpy of feedwater
w_{dc} = downcomer flow rate
h_{D2} = lower drum (node 2) liquid enthalpy

Downcomer. The temperature of a given particle of water does not change as it passes through the downcomer; therefore, the downcomer can be described as a pipe with a constant transport delay when the flow rate is constant. The delay is simulated by an inverter and an integrator with the following transfer function:

$$\frac{\Theta_{dc}}{\Theta_{D_2}} = \frac{1 - (T_D/2)p}{1 + (T_D/2)p} \quad (3)$$

where

Θ_{dc} = outlet temperature of downcomer
T_D = pipe transport time (sec)
p = operator (d/dt) (sec^{-1})

Heat exchanger and riser. The short transport time of the two-phase flow in the riser allows it to be considered as a part of the heat exchanger. Only saturated liquid is considered in actual contact with the boiler tubes on the steam side, and the heat capacity of the vapor is neglected. The heat-exchanger energy balance is

$$d(\gamma M_{HX}\Theta_{HX})/dt = UA(\Theta_{pc}^{avg} - \Theta_{HX}) + w_{dc}h_{dc} - w_{rf}h_{rf} - w_{rg}h_g \quad (4)$$

where

M_{HX} = heat exchanger liquid mass
Θ_{HX} = heat exchanger saturation temperature
UA = heat transfer coefficient across boiler tubes (Btu/sec-° F)
Θ_{pc}^{avg} = average primary coolant temperature in boiler tubes
h_{dc} = downcomer outlet enthalpy
w_{rg} = riser vapor flow rate

If θ_{D1} attempts to exceed θ_{HX}, w_{Dfg} assumes a variable value that keeps θ_{HX} equal to θ_{D1}. When θ_{HX} is greater than θ_{D1}, no flashing occurs in the steam drum, thus making w_{Dfg} equal to zero.

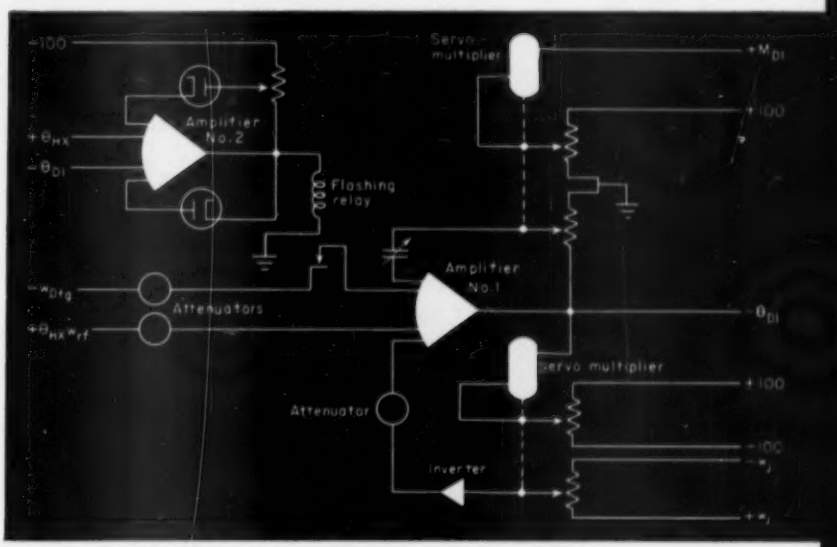


FIG. 2. Computer circuit for upper-drum temperature representation

Boiler Mass Balance

In the model, the transient heat-exchanger mass M_{HX} is composed of the following terms: (a) steady-state swell of the heat exchanger from boiler load changes; (b) deviations of downcomer subcooling from its steady-state value; (c) deviation of boiler pressure from its steady-state value; and (d) excessive boiler-tube heat flux.

A mass balance on the entire boiler will determine the effective heat capacities at any particular time. The mass balance can be written

$$M_{HX} + M_{D1} + M_{D2} + M_{dc} = M_{initial} + \int_0^t (w_{fw} - w_s)dt$$

where

M_{dc} = downcomer mass
w_s = steam flow out of boiler

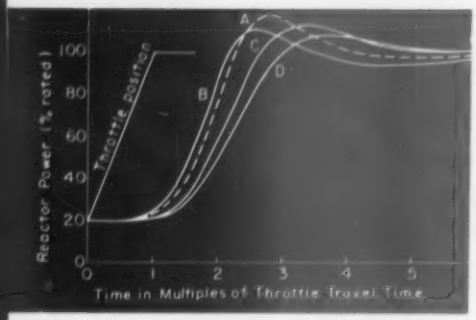


FIG. 3. Response of power to load increase. Curve A is for one node; B, C, and D, four node. No upper drum flashing in curves A and B. Feedwater lags steam flow in curve D

The flows w_i and w_{ef} are obtained from the mass balances on the heat exchanger and lower drum nodes. The steam flow delivered by the boiler to the load is

$$w_s = w_{re} + w_{Dfe}$$

The enthalpy of liquid is assumed proportional to its temperature. The boiler's heat-transfer coefficient UA is varied either as a function of steam flow or of the heat flux.

Analog Representation

The techniques used to solve ordinary *linear* differential equations with constant coefficients are described adequately elsewhere (*1, 2*). Each type of *nonlinearity* appearing in an equation requires a different technique, however. Handling the nonlinearities of Eq. (1) requires: (a) the solution of an equation containing the rate of change of the product of two variables; and (b) a representation of the flashing of water into steam in the upper drum to maintain drum temperature equal to or less than heat-exchanger temperature.

Time rate of change of $M_{D1}\Theta_{D1}$. The computer circuit used to obtain the solution of Eq. (1) is shown in Fig. 2. Servomultiplier No. 1 will move the arms of the mechanically ganged potentiometers to a position proportional to the input voltage M_{D1}. The capacitor between the arm of the potentiometer and the summing point of amplifier No. 1 enables the amplifier to solve differential equations of the type

$$d(\beta e_{out})/dt = -\Sigma K_i e_i$$

where

β = ratio of output voltage to input voltage of the servo-driven feedback potentiometer
e_{out} = amplifier output voltage
K_i = gain applied to ith input voltage
e_i = ith input voltage

The term β represents the normalized or per-unit value of the upper drum mass in this instance. The capacitor is connected to the output of amplifier No. 1 when $\beta = 1$ and the circuit operates as a conventional integrator. If β is made to vary as the normalized upper drum mass M_{D1}/M_{D1m} by means of the servomultiplier, the circuit will determine the temperature of a variable mass region when the next gain or loss of energy of the node is known.

The time constant of the feedback capacitor and the potentiometer resistance represents an error in the solution of the differential equation. The error, however, is small if the problem is scaled so that the maximum rates of change of Θ_{D1} and M_{D1} are less than a few volts per second.

Drum-flashing representation. The water in the upper steam drum begins flashing into steam whenever the drum temperature exceeds the heat-exchanger temperature. In the analog circuit of Fig. 2, amplifier No. 2 monitors the difference between Θ_{D1} and Θ_{HX}. As long as Θ_{D1} is less than Θ_{HX}, no flashing occurs and the flashing relay remains open. When Θ_{D1} exceeds Θ_{HX}, the amplifier closes the flashing relay. Whenever this relay is closed, a large constant flashing steam rate w_{Dfe} is considered to flow from the upper drum. The value of w_{Dfe} is chosen large enough to keep Θ_{D1} less than Θ_{HX} even for the most severe drop in Θ_{HX} that could be experienced.

Contacts of the flashing relay are used to preserve mass and energy balances while flashing is in progress. The contacts in Fig. 2 represent the term $w_{Dfe}h_e$, the energy loss of the upper drum caused by the flashing.

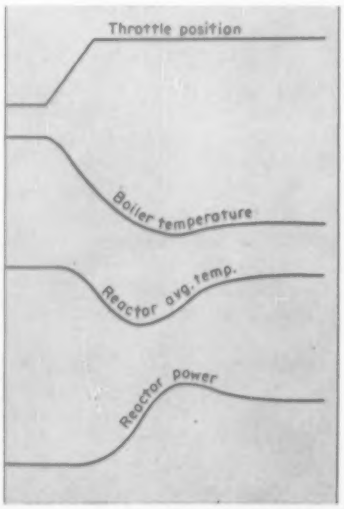
The contacts of the flashing relay open when Θ_{HX} again becomes larger than Θ_{D1}. If the load on the heat exchanger is sufficient to produce more flashing, the relay will cycle. Averaging the pulses produced by the flashing relay over a period of time will approximate the actual flashing rate.

Control by Temperature Coefficient

Power plant control by reactor response to steam demand (*3*) is illustrated by the curves at the right which show plant response to a load increase imposed by opening the turbine throttle valve.

As the throttle valve is opened, the heat flux from the primary coolant to the boiler becomes insufficient to supply the energy of the higher steam flow. The additional energy now required is supplied by the boiler heat capacity, with a resultant drop in boiler temperature. The increased temperature differential between the boiler and the primary coolant flowing through it causes an increase in the amount of energy that is removed from the coolant, with a resultant drop in coolant temperature at the boiler outlet.

The cold-leg temperature wave reaches the reactor after a delay corresponding to the cold-leg transport time. Now the average reactor temperature begins to follow the boiler temperature, and the reactor power level finally begins responding to the load change. Since the boiler and the reactor have been cooled below their final steady-state values by the load change, the reactor must ultimately supply all the energy required by the load and also return the plant temperatures to their final steady-state values. This is accomplished by the overshoot in reactor power.



Results

The effects of steam demand and feedwater stability on reactor operation as revealed by the analog study follow.

Load disturbances. The effects of changing throttle-valve position on reactor power level are illustrated in Fig. 3. Throttle-valve positions are given in the percentage of rated steady-state steam flow provided by each position from 20 to 100%.

In the first three curves, feedwater flow equals steam flow. In curve D, feedwater flow was forced to lag behind steam flow with a delay time of 10 sec followed by a time constant of 10 sec. The feedwater flow of curve D is considered typical of what might be obtained with a practical three-element feedwater control system.

Two major differences in reactor power behavior between the one- and four-node steam-side representations become apparent upon inspection of Fig. 3: (a) for all three variations of the four-node model, the peak reactor power is less than with the one-node model; (b) the peak neutron flux of curve D, which is considered the most probable plant response, occurs about 0.75 time units after the peak of the one-node model. Both of the foregoing effects are in a direction which might be expected if one were to determine the transients experimentally.

Two reasons for the differences between curves A and D exist. The first reason is that for the same steam-side heat capacity and the same drop in heat-exchanger temperature between 20 and 100% rated output, the four-node model will provide more stored energy than the one-node model. This extra energy arises because of the subcooling of the lower drum and downcomer.

With a constant downcomer flow of five times rated steam flow and feedwater subcooling in the region of 200–250° F, the lower drum and downcomer subcooling will increase about 32° F between 20 and 100% rated steam flow.

The additional stored energy thus available in a maneuver becomes greater with larger subcooled regions in the steam drum and downcomer. This extra stored energy moderates the rate of steam temperature decay for a given load increase. Consequently, the reactor average temperature also drops at a slower rate.

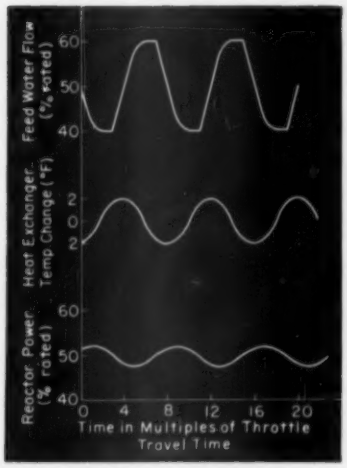


FIG. 4. Effect of feedwater flow on reactor power level

The second reason why curves A and D of Fig. 3 differ is that subcooled feedwater tends to chill the lower drum and downcomer. If the feedwater increase is delayed during the load change, the amount of feedwater that must be heated to system temperature is reduced. Even though the available mass of the steam side is also reduced, the rate of steam temperature decrease is again moderated because less of the boiler output is being used for feedwater heating. This moderating effect ceases if the feedwater delay becomes so great that the natural circulation is destroyed.

In discussing the reasons for the differences between curves A and D of Fig. 3, it is well to mention the steam-side variations which did *not* have a significant effect on reactor power. Two such variations are: (a) Transient disturbances in downcomer flow did not materially affect reactor power level. The downcomer flow was arbitrarily varied from its steady-state value for a few seconds by as much as a factor of two with negligible effect on reactor power. (b) Boiler swell, the transfer of extra water from the heat exchanger to the steam drum, also had little effect on reactor power level. The flashing process, which keeps the upper drum and steam temperatures together as heat exchanger temperature falls, undoubtedly helps to minimize the effect of swell on neutron flux.

During the power transients, the steam and main coolant temperatures of the four-node model also vary at a slower rate than do those of the one-node model. However, the maximum swings in temperature were about the same in both models; therefore, the maximum volume surge from the primary expansion tank was also about the same but occurred more slowly in the case of the four-node model.

Feedwater stability. Figure 4 illustrates the effect on reactor power level of slow feedwater oscillations of about 10% rated feed flow with the turbine throttle remaining in its 50% load position.

The heat-exchanger temperature oscillation is shown as having an amplitude of about 2° F. This oscillation in steam temperature is transmitted to the primary coolant, where it finally produces the reactor power oscillation having an amplitude of about 2.5%. These results indicate that if an automatic reactor power-level control loop were incorporated to assist the natural plant temperature coefficient, long-term feedwater oscillations might cause undesirable cycling of the power-level controls.

For a given frequency and magnitude of feedwater oscillation, the swing in downcomer temperature and, ultimately, the reactor power oscillation can be reduced by increasing the effective heat capacity of the lower drum. This benefit must be weighed against a resultant increased tendency toward downcomer flashing whenever the load is increased.

Conclusions

The one-node model provides results which are conservative so far as the behavior of reactor power is concerned. Therefore, a one-node model would generally be sufficient for preliminary investigations of reactor power during load disturbances. In addition, the one-node model would suffice for almost all investigations where (a) the reactor temperature coefficient is negligible and the steam temperature is required merely as a base for primary coolant temperatures, or (b) where the boiler is so designed that the circulation time of the steam side is small compared with that of the primary coolant loop.

If more exact values for reactor power overshoots, reactor period and primary coolant surge rate are desired for protective-system and pressurizer design, refinement of the steam-side model becomes just as important as improvement of the reactor and primary coolant representations. For ex-

ample, in Fig. 3 the differences in reactor power behavior produced by changes in the steam-side representation were more significant than those produced either by changes of 50% in the hot- and cold-leg transport times or by a more elaborate representation for reactor residual heat.

The four-node steam-side representation is recommended as a reasonable compromise in cases where a detailed investigation of the natural plant characteristics is required but the amount of available analog equipment is limited. The four-node steam-side model presented is certainly incomplete—the most serious limitation probably being the neglect of the natural-circulation equations and the assumption of constant downcomer flow.

However, if the limitations are recognized and the computer is monitored for conditions that invalidate the assumptions, e.g., downcomer flashing in the case of constant downcomer flow, the four-node steam side will provide results significantly closer to physical reality than those of the one-node model.

* * *

This article was prepared as a result of work performed under AEC Contract W-31-109-Eng-52.

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3. M. A. Schultz, "Control of Nuclear Reactors and Power Plants" (McGraw-Hill Book Co., Inc., New York, 1955)

A Nucleonics Survey: Analog Computer Use in the Nuclear Field

ANALOG COMPUTERS used in the nuclear field seem to be as varied as the applications to which they are put. All the major manufactures are represented in this survey of users in the nuclear field conducted by NUCLEONICS. However the survey is not as extensive as that for digital machines on p. 69.

Where Analog Computers Are Used in the Nuclear Field

User organization	*Computers used*	*Outside machines used*	*Future plans*
Alco Products		Yes	Yes
Allis Chalmers	Philbrick		
American Machine & Foundry	Midcentury		
Combustion Engineering	PACE	REAC	
Curtiss-Wright	REAC "400"		Double capacity
General Electric (ANP, Lockland, O.)	GEDA; PACE 16-31R, and 24-D		
General Electric (KAPL)	PACE		
General Motors	EASE		
Hanford Atomic Products Op.	GEDA A-14		
Lockheed Missile Systems Div.	16-31R PACE		
NACA	Passive simulators*		
Oak Ridge National Lab.	"Home-made"		
Phillips Petroleum Co.			Reactor simulator
Westinghouse Bettis Plant	16-24, 16-131R, 16-31R PACE, EASE		

* D. Fiene, H. Schneider, R. B. Spooner NUCLEONICS **11**, No. 8, 16 (1953).

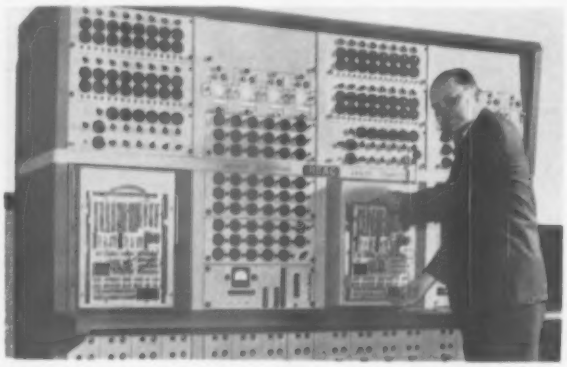


FIG. 1. Reeves 400 REAC computer as supplied to the French Atomic Energy Commission

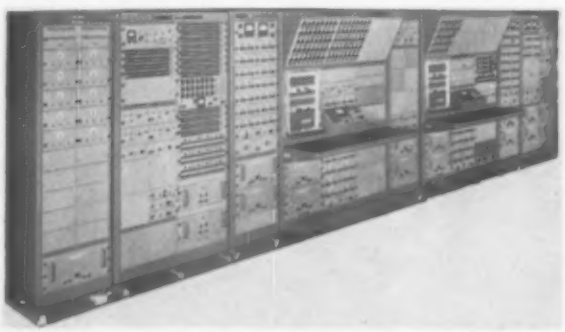
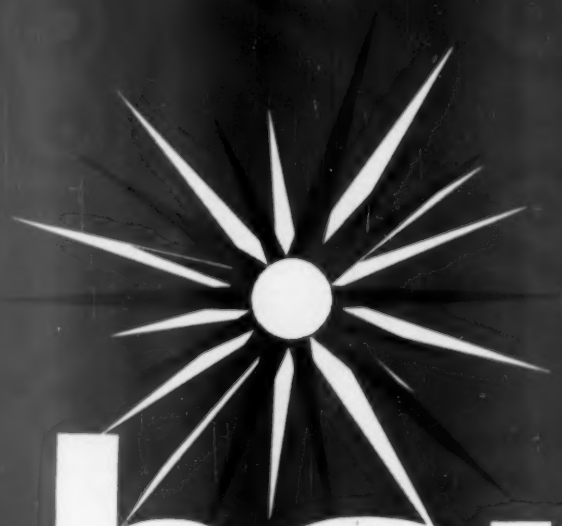


FIG. 2. Upper panels of advanced analog computer by Electronic Associates, Inc.



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Low-Background Counter for Solid β-Emitting Samples

By S. D. SOFTKY* and R. E. NATHER*
Tracerlab, Inc.
Western Division
Richmond, California

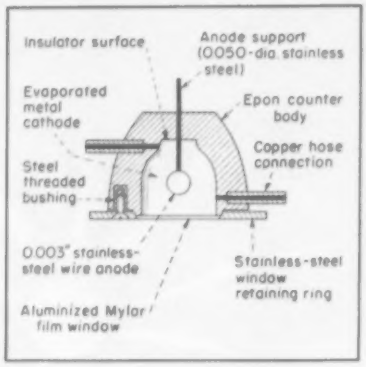


FIG. 1. Epon sample counter

The common approach to counting beta activities of a few counts per minute is to have a central sample counter in anticoincidence with a surrounding ring of G-M counters to eliminate cosmic-ray background (*1, 2*). The assembly is placed inside a shield to reduce local background activity. Although the counting arrangement described here follows the same approach, a new counter material and anticoincidence-ring design offer advantages for routine use.

Central Counter

The central sample counter was made of Epon resin† because of its excellent insulating properties and freedom from natural radioactivity. The counter (Fig. 1) has a 1.310-in.-diameter window and a jug-shaped sensitive volume 1.25 in. deep to allow using 1-in.-diameter sample planchets.

The anode support is located in the neck of the jug to expose a minimum of insulator surface to the sensitive counting volume. The 0.9-mg/cm² window is 0.00025-in.-thick Mylar film, aluminum-vapor-coated on both sides to render it conducting.‡ The anode wire is a loop of 0.003-in. stainless-steel wire, specially drawn to ensure uniform diameter, supported by a length of steel syringe tubing. The solder joint is made outside the counter.

A stainless-steel ring, beveled on the inner surface, holds the window in place and serves to place the window material under tension during assembly. Two $\frac{3}{16}$-in.-o.d. copper pipes are cast into the counter to allow gas flow through the sensitive volume.

The counter is completely cast in a single operation, requiring no machine work. A special mold allows the anode support, the gas pipes, and the threaded copper bushings that hold the window ring to be cast into the counter at the time it is made. Stainless steel is vacuum-evaporated onto the inner surface of the counter to form the cathode, after which the anode and window are installed.

The first Epon counters exhibited a spurious counting effect, which was particularly noticeable after exposure to a high-activity sample. This "memory effect" was apparently due to the buildup of static charge on the exposed portions of the insulator. Some photosensitivity was also observed. The addition of lampblack to the Epon solved both problems by lowering the resistivity of the insulating material and rendering the counter opaque to visible light.

The counters can be operated either as proportional counters (using methane flow gas) or as G-M counters (using helium-butane or helium-isobutane). The latter mode of operation is chosen in this case, since the larger signal available from a G-M counter greatly simplifies the electronics required, and the

* PRESENT ADDRESS: Aerojet-General Nucleonics, San Ramon, California.

† Epon resin type 828, Shell Chemical Co.

‡ Minnesota Mining and Manufacturing Co.; available from Wakefield Industries, Skokie, Ill., and Hastings and Co., Philadelphia, Pa.

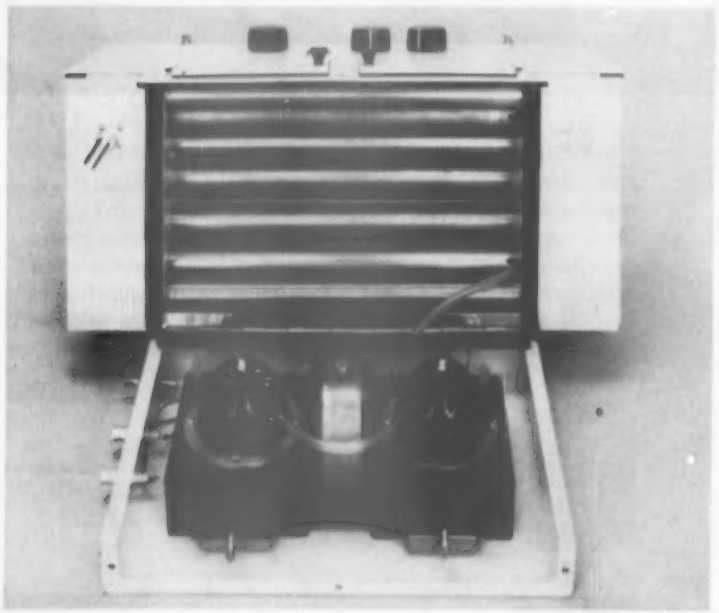
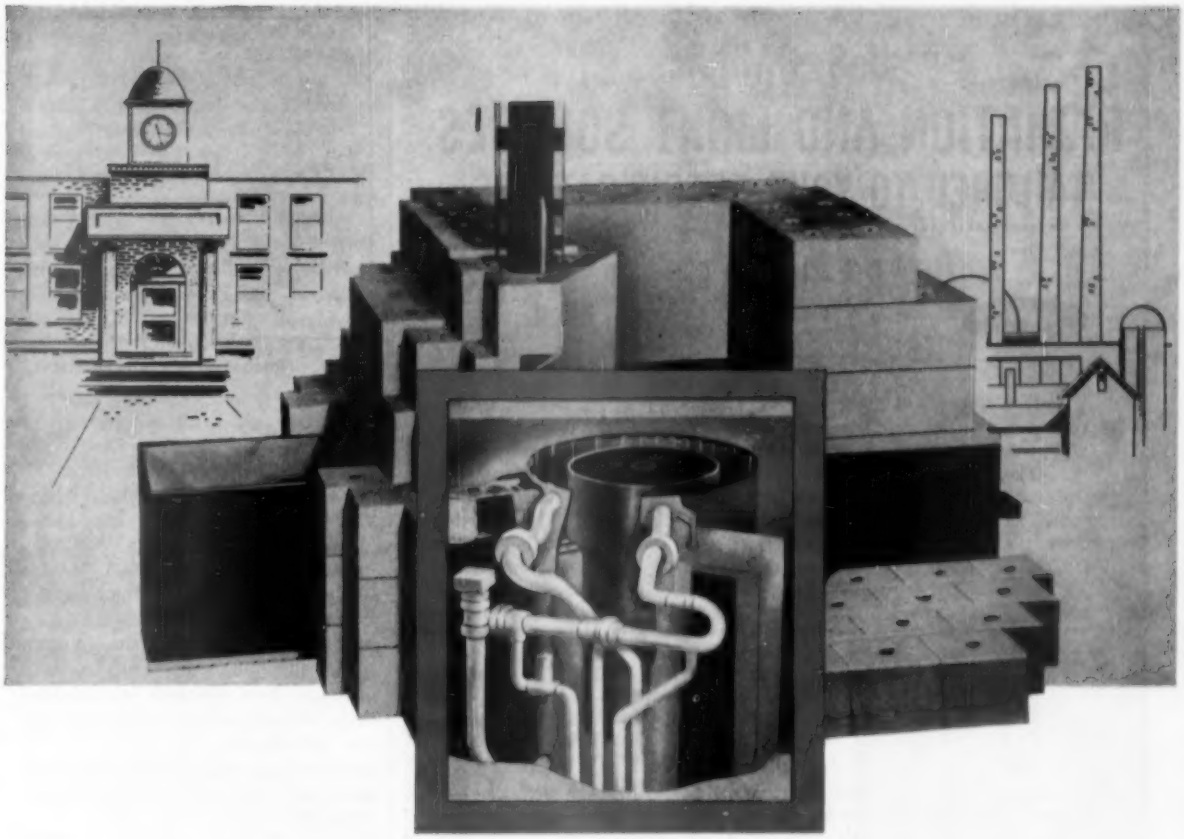


FIG. 2. Counting assembly with steel dust cover open. Anticoincidence umbrella is in top and two sample counters at bottom separated by mercury shield



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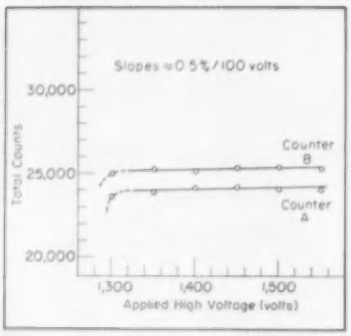

FIG. 3. Plateaus of two Epon counters

long deadtime is a negligible problem at low counting rates. Helium is also less sensitive to gamma radiation than the heavier methane.

The support and sample-slide mechanism are also cast of Epon resin to minimize natural radioactivity directly below the window. The sample-slide height is adjustable by means of a screw so that effective geometry can be standardized. Two counters are mounted under a single anticoincidence umbrella and are separated by a 1-in. wall of mercury in an Epon bottle (Fig. 2) to minimize "cross talk" between the two counters.

Anticoincidence Ring

After two unsuccessful attempts to construct a single, multiple-wire anticoincidence umbrella, it was decided to use several G-M tubes as the shielding counter.

A complete ring of tubes around the central counters was not used because there would be gaps between tubes and sample changing would be difficult. A "folded-ring" system was chosen—a semicircular umbrella with additional tubes covering the gaps. This allows easy sample changing.

Eleven G-M tubes§ are used in the umbrella. All the tubes are connected in parallel electrically for normal operation. An 11-position switch is provided so that each tube may be counted by itself to locate a faulty tube.

The over-all plateau of the eleven tubes in parallel was remarkably flat, although not quite so long as that of an individual tube because of variations in starting voltage. Specifically, the plateau was 200 volts long, with a slope of less than 2% per 100 volts. It should be noted, however, that a plateau as such is not required for the umbrella, the only requirement being

§ Type TGC-16, Tracerlab, Inc., Boston, Mass.

that all of the ionizing cosmic rays be counted. A more meaningful plateau would result from measuring the central counter background as the voltage is changed on the umbrella.

Shield, Scaler

The 8-in.-thick shield is made of mild-steel bricks (4 × 4 × 8 in.)—lead usually contains a prohibitive amount of natural radioactivity and emits troublesome gamma-excited fluorescent radiation. Total shield weight is 5,000 lb.

Two of the scaler channels record central-counter pulses less coincident umbrella pulses. A selector switch for the third channel permits counting umbrella pulses or central-counter pulses before coincident umbrella pulses are introduced. The latter measurements are valuable to determine if a high central-counter background is due to contamination or electronic difficulties.

Normal operating voltages are ~1,100 volts for the umbrella and 1,300–1,400 volts for the central counters.

Counter Characteristics

The plateaus of two counters are shown in Fig. 3. Gas flow is held at 20 cm³/min. Plateau characteristics deteriorate rapidly at flow rates less than this.

The gamma sensitivity of the central counters was measured with a Zn^{65} source (calibrated at 3×10^5 gammas/min) covered with a copper absorber to stop betas. The observed counting rate of 81 cpm indicates a gamma sensitivity of ~0.1%.

The efficiency of the mercury shield between the central counters was tested with an Fe^{59} source in one counter. The latter counter showed a counting rate of 917 cpm, while the other showed 0.65 ± 0.06 cpm. With the source removed, the second counter showed 0.63 ± 0.06 cpm, indicating the shield is satisfactory.

A test with a 1-mc Co^{60} source on top of the steel shield indicated that the maximum gamma activity allowable in the vicinity of the counting assembly is 1 μc.

The standard deviation of the backgrounds observed on this counting assembly is 0.01 cpm. The limit of sensitivity is then considered to be 0.05 cpm.

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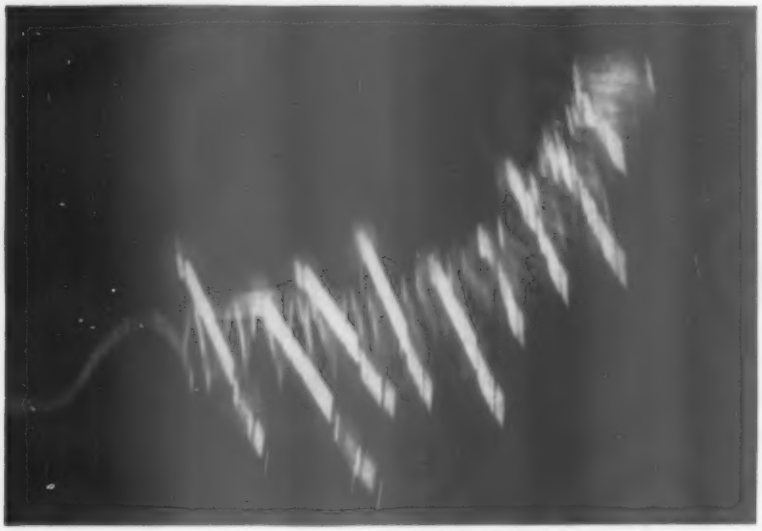

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Rare-Earth Solutions for 4π Counting

By JUAN PASCUAL and
EDWARD C. FREILING

Nuclear and Physical Chemistry Branch
U. S. Naval Radiological Defense Lab.
San Francisco, California

To determine rare-earth absolute fission yields accurately, solid-free solutions of rare-earth activities must be prepared for absolute-disintegration-rate determination by 4π counting. The method described here is equally applicable to the preparation of lanthanide, actinide, and yttrium solutions. With minor modifications it is suitable for preparing alkaline-earth solutions. It can also be used to recover rare-earth fission products heavier than Tb from lactate eluants in good yields whether or not carriers for these activities are present.

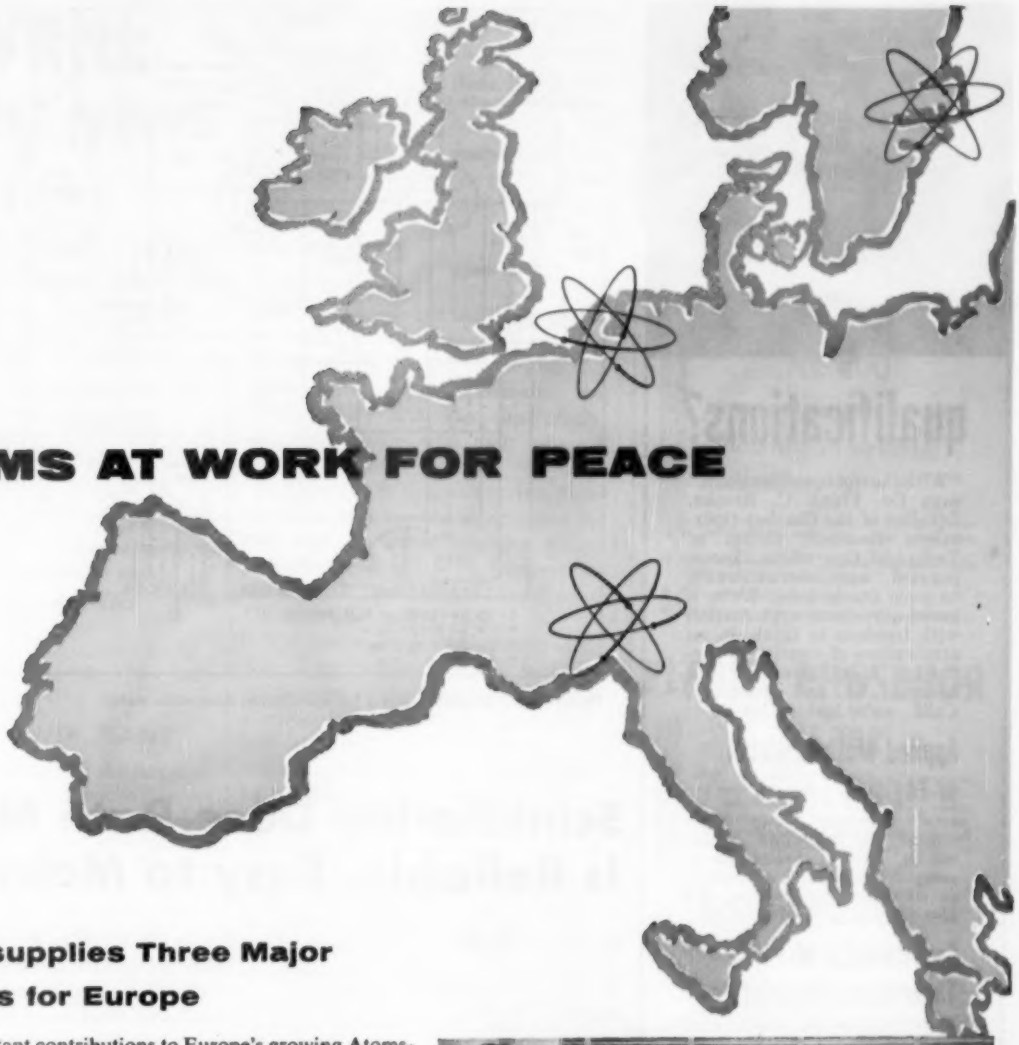
Procedure

Two milligrams of La carrier is added to an HCl solution of uranium and fission products. The rare-earth group is purified.* The individual rare earths are separated in a 2-mm × 60-cm column of Dowex-50 in the NH_4^+ form at 87° C, the lactic acid gradient elution conditions described by Bunney et al. being scaled down to these column dimensions.

Those fractions of effluent that contain the activity of interest are combined and acidified with HCl to *p*H 2. The acidified solution is extracted four times with equal volumes of diethyl ether; with each extraction it is mechanically stirred for 1 min. This extraction reduces the concentration of lactic acid to such an extent that the rare-earth activity can be adsorbed on passing the raffinate through a 2-mm by 10-cm column of Dowex-50 at room temperature. Washing with four column volumes of deionized water frees the column of dissolved solids. The rare-earth activity is eluted with 6*N* HCl, and finally the fraction of effluent containing the activity is digested with nitric acid to remove HCl and remaining traces of ammonium ion.

Dilution of the nitric acid to a strength of 1*M* yields a solution with 0.15 mg of solid or less per milliliter and well within the range of 4π counting.

* L. R. Bunney, E. C. Freiling, L. D. McIsaac, E. M. Scadden, NUCLEONICS **15**, No. 2, 81 (1957).



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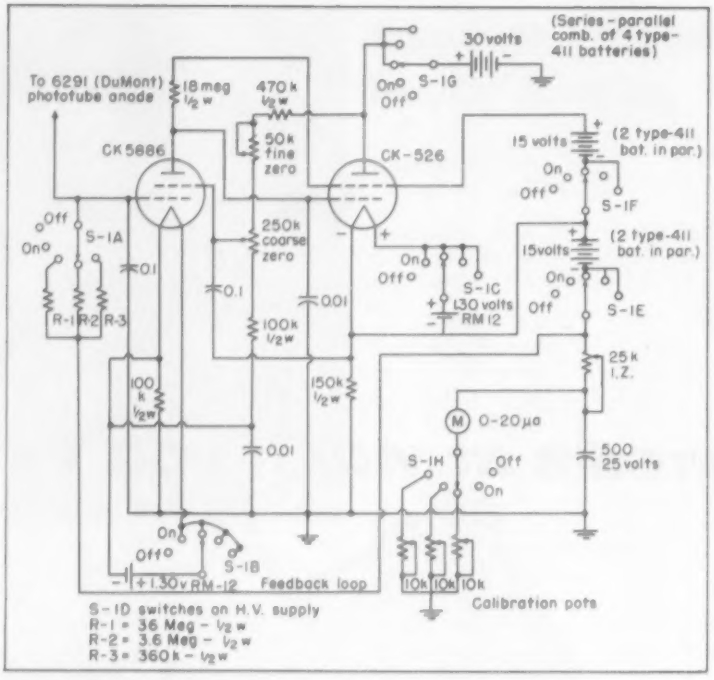

FIG. 1. D-c amplifier circuit for scintillation dose-rate meter

Scintillation Dose-Rate Meter Is Reliable, Easy to Maintain

By W. G. SPEAR
Biophysics Section
Hanford Atomic Products Operation
Richland, Washington

Portable exposure-rate instruments for beta-gamma measurements generally use an ion-chamber detector and a simple d-c amplifier circuit utilizing an electrometer tube and input resistors of $\sim 10^{12}$ ohms. Such systems, although reasonably reliable, are subject to error and failure because of the exceedingly low available chamber current.

The portable scintillation beta-gamma exposure-rate meter described

FIG. 2. Ratemeter with cover removed; light shield is at right

WKNL IN THE NEWS

NUCLEAR BATTERY

Walter Kidde Nuclear Laboratories and Elgin National Watch Company recently announced the development of a miniature atomic battery capable of delivering continuous electrical power for at least five years.

The battery, which utilizes the decay of a beta-emitting radioisotope as its source of energy, is completely safe for extended personal use. Not yet commercially available, the battery may eventually be used in hearing aids, transistorized radios and wrist watches. Its resistance to temperature extremes promises additional use in missile and rocket work and in scientific deep space experiments.

★ ★ ★

NUCLEAR POWER PLANT

WKNL has completed the design of a one-cycle emergency nuclear power plant for the Corps of Engineers, Research and Development Laboratories, Fort Belvoir, Va. Details of the work are classified.

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The portable gamma-irradiation facility designed by WKNL for the B. F. Goodrich Research Center at Brecksville, O., was included in a recent report on significant Cobalt 60 facilities. Currently in its second year of operation, the lead pig was designed to meet the following Goodrich requirements: safety, maximum irradiation volume in several spaces, weight limit of 5 tons, operational flexibility, uniform flux, reliability and readily changed sources.

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here employs a DuMont 6291 photomultiplier tube and a thin crushed anthracene crystal weighing 200 mg/cm². The instrument has three ranges: 0–10, 0–100, and 0–1,000 mr/hr. The crystal and light shield are thin to permit good beta response at the lower energies. The device weighs 5½ lb.

Discussion

The meter was designed for good stability of operation and low maintenance cost. The use of a high-gain d-c amplifier with feedback following the photomultiplier tube eliminated the necessity for critically clean high-megohm resistors with their inherent leakage problems.

A standard 36-megohm carbon resistor is employed as the input-signal resistor on the most sensitive range. This size requires 7×10^{-9} amperes to obtain the 0.25 volt needed to drive the CK-5886 electrometer tube.

The time constant of previous meters was 15 sec or more, whereas with the scintillation system the time constant is reduced to 5.5 sec on the most sensitive range. A further reduction in time constant could be accomplished by reducing the 0.1-μf shunt-capacitor value; however, this reduction would decrease the instrument circuit damping.

In the d-c amplifier circuit* (Fig. 1), the electrometer tube is coupled directly to a pentode cathode follower, which is arranged to match the meter impedance for optimum power transfer. Voltage feedback is applied to the input grid by means of *R*-1, *R*-2 or *R*-3, depending upon which range is used. This allows the signal current to change considerably without greatly affecting the input-grid voltage. The plate voltage supply for the electrometer tube is returned to the cathode of the cathode follower to increase the effective load impedance of the electrometer tube, and hence increase the potential gain of the circuit. The proper phase relationship for feedback is retained.

A CK-5886 was used for the electrometer tube since it has a transconductance of nearly 1,000 μmhos. To keep the grid current as low as possible, the plate voltage is held to about 15 volts with 1 μa of plate current flowing. This allows the use of small hearing-aid batteries since the combined quiescent direct current for the electrometer tube, electrometer-screen voltage divider and cathode-follower plate current is

* N. F. Moody, *Rev. Sci. Instr.* **22** (1951).

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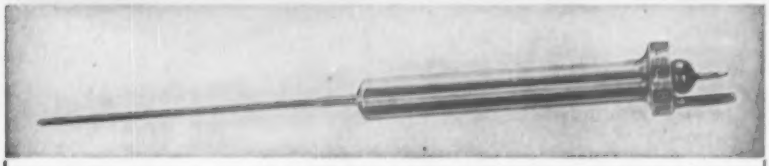

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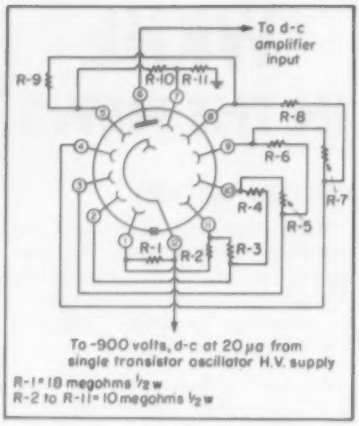

FIG. 3. Wiring of DuMont 6291 photomultiplier

only about 120 μa. Zeroing the instrument is accomplished by varying the screen-grid voltage of the electrometer tube. This is preferred to varying the control-grid bias, since the grid current is quite dependent upon grid bias, and it is desirable to hold this bias within narrow limits.

The 100,000-ohm resistor in the cathode circuit of the electrometer tube may be varied by ±10% if necessary when the electrometer tube is replaced to place the new tube at its correct operating point. Three range-adjustment rheostats are available externally for calibration.

Sensitivity. Varying the 25,000-ohm rheostat in Fig. 1 varies the response time and sensitivity of the instrument, with the undesirable feature of decreasing zero stability with increased sensitivity.

For measurements not exceeding 15 or 20 min, the sensitivity may be increased to 1 or 2 mr/hr full scale on the first range with assurance that the zeroing will be stabilized for that short period. Adjustment of the rheostat to give <5% error in zero stability over a 4- or 5-hr period results in a maximum sensitivity of about 10 mr/hr full scale on the most sensitive range.

Zero-adjusting. The instrument must be zero-adjusted on each range since the constant photomultiplier dark current through the three different range resistors gives different grid voltages and, hence, different meter readings. Since the grid resistors are in ratios of 10, i.e., 36 and 0.36 meg, the variations in meter zero adjustment when changing ranges are appreciable.

When switching ranges, a light shield consisting of a brass slide in a thin aluminum housing, as shown in Fig.

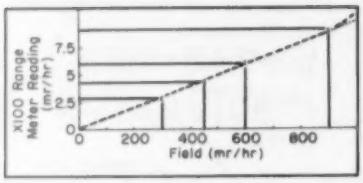

FIG. 4. Ratemeter calibration curves obtained with radium sources

2, is placed between the anthracene crystal and the photomultiplier. This method allows the photomultiplier-dark-current meter reading to be zero-adjusted under any field conditions since light photons being emitted by the crystal are not striking the photocathode.

Insertion of the light shield is accomplished by inverting the instrument so that gravity pulls the shield down. With the shield in place, the instrument may be rotated to a horizontal position and then zero-adjusted. After adjusting, returning the instrument to the normal vertical position restores the shield and the instrument is ready for operation.

Slide sticking is minimized by using an aluminum housing and a brass shield. The complete operation of switching ranges and adjusting can be accomplished in 15–20 sec.

Since the grid-input resistors are of relatively low values compared with grid resistors in ion-chamber instruments, the precautions of careful cleaning, insulator selection, switch selection, and grounding of all resistors except the one in use are not necessary. This lowers costs too.

High-voltage supply. The high-voltage supply is a single-transistor supply.†

The supply, powered by mercury cells with a battery life of 300–400 hr, delivers a negative 900–1,000 volts at 20 μa to the photomultiplier cathode. The negative high voltage is used so that the tenth dynode of the photomultiplier is grounded through an 18-meg resistor with no high voltage applied to the anode. As a result, the photomultiplier anode can be coupled directly to the electrometer-tube grid.

Figure 3 shows the photomultiplier wiring schematic diagram.

Results and Conclusions

The instrument-calibration curves obtained with radium sources at vari-

† W. G. Spear, HW-38682 (1955).

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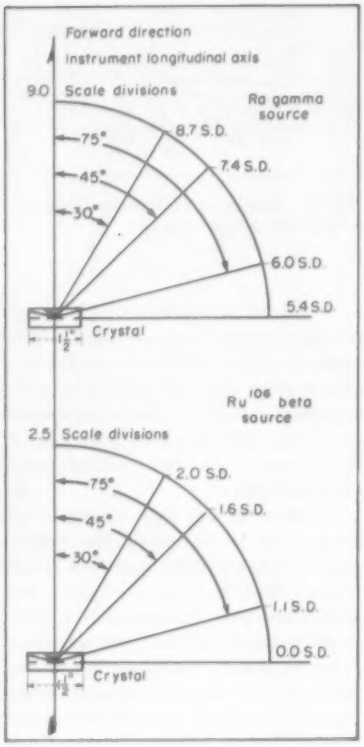

FIG. 5. Ratemeter angular-response curves. Sources are essentially point sources. Measurements were made on 5-in. radius. Readings were on 0–10-mr/hr range with full scale of 10 divisions

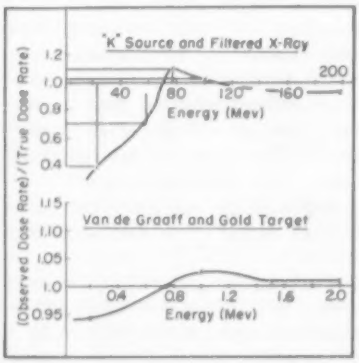

FIG. 6. Ratemeter energy-dependence curves

ous distances showed good linearity. Figure 4 shows the curve for the 0–1,000-mr/hr range; the other ranges had as good linearity. The instrument reading versus point-source-position curves are shown in Fig. 5. The battery life of the instrument is ~200 hr of continuous use.

Temperature-dependence tests from −20° F to +160° F revealed no faults, although a slight zero-setting change was noted because of the change in dark current of the photomultiplier

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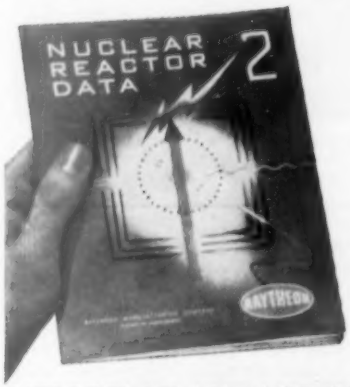

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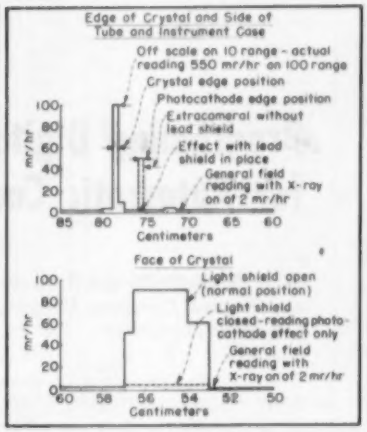

Edge of Crystal and Side of Tube and Instrument Case
Off scale on 10 range - actual reading 550 mr/hr on 100 range
Crystal edge position
Photocathode edge position
Extracameral without lead shield
Effect with lead shield in place
General field reading with X-ray on of 2 mr/hr
mr/hr
Centimeters
Face of Crystal
Light shield open (normal position)
Light shield closed - reading photocathode effect only
General field reading with X-ray on of 2 mr/hr
mr/hr
Centimeters

FIG. 7. Extracameral-effect scanning curves obtained with 200-kev X-rays and 2-mm slit in lead shield 50 ft from X-ray tube; both curves were measured on 0–100-mr/hr range. Side scan was made with ⅛-in.-thick ¾-in.-wide lead strip shielding edge of photocathode

at the various temperatures involved. This problem is resolved by rezeroing the instrument if the temperature changes markedly.

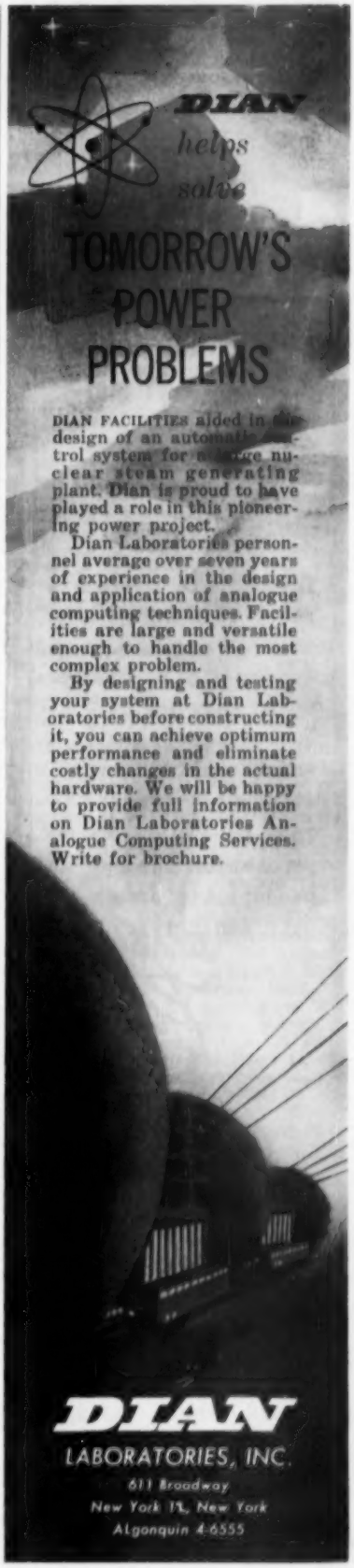
Figure 6 shows the energy-dependence curves. The first curve from 23 kev to 100 kev was obtained with various K X-ray sources, and the 200-kev point was obtained using a filtered X-ray. The second curve shows the results from 200 kev to 2.0 Mev with the points from 750 kev to 2.0 Mev obtained by using a Van de Graaff accelerator and a gold target.

The two curves in Fig. 7 show the crystal response and the extracameral effect from the photocathode obtained by using the 200-kev X-ray and a 2-mm-wide slit in a lead shield 50 ft from the X-ray tube. It is apparent from curve A that the ⅛-in. lead shield around the photomultiplier photocathode is necessary to reduce to a negligible amount the extracameral response, for the particular field strength may be reduced to zero on the meter.

The use of a 200-mg/cm² crushed anthracene crystal and a 15-mg/cm² lighttight covering of Mylar and black optical tape over the crystal resulted in a reading ratio of 2:1 with the crystal ⅛ in. from a calibrated uranium slab. The ratio is obtained by dividing the true dose rate from the slab by the reading obtained.

* * *

The author wishes to acknowledge the aid of L. V. Zuerner in obtaining the test data. This work was performed for the U. S. Atomic Energy Commission under contract W-31-109-Eng-52.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Programing Digital Register for Automatic Counting Plots

By D. S. HARMER and R. A. JAMES

Department of Chemistry, University of California
Los Angeles, California

In many counting situations where accuracy is desired, the data are either recorded by hand or on tape (*1*) and then plotted by hand. If several runs are made, the labor and time consumed in data plotting become quite large. To avoid these disadvantages, readout is frequently provided by a counting-rate meter and recorder with attendant sacrifice of accuracy.

The programing digital reader (PDR) described here eliminates these disadvantages and combines the advantages of the printing register and the counting-rate-meter methods.

With the PDR, the counting interval can be set long enough to average out the statistical fluctuations, and the counts can be plotted directly by a two-axis graphic recorder. The count and readout circuit used in the PDR eliminates the drift and calibration problems of most electronic readout devices. The circuit has a stability and accuracy of readout equivalent to hand-plotting the data on a similar scale.

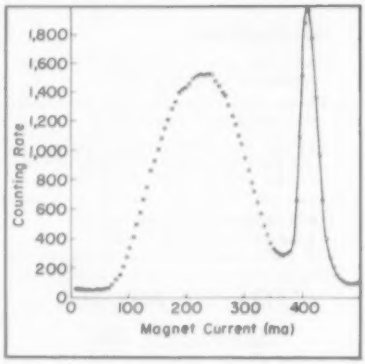

FIG. 1. Beta spectrum of Cs^{137} using PDR and low-resolution spectrometer; dots printed by recorder are circled

Since the counting time, sequence, and duration of opening and closing auxiliary circuits can be varied easily to suit the needs of a particular counting situation, the instrument also can be used in many other readout or pro-

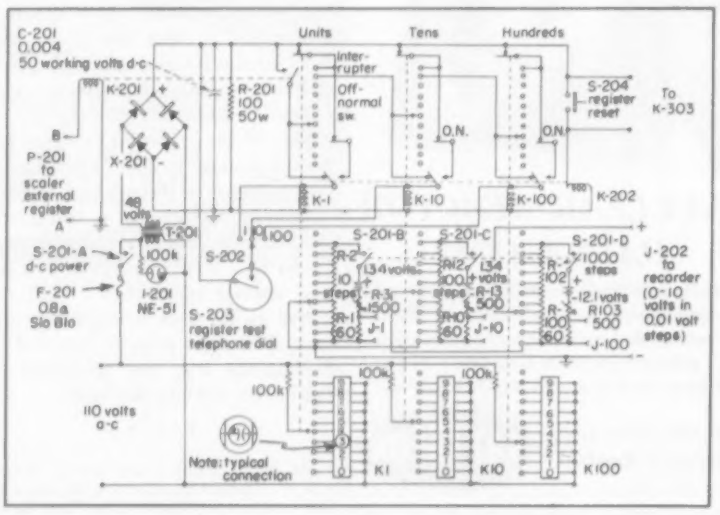

FIG. 2. Register section of PDR

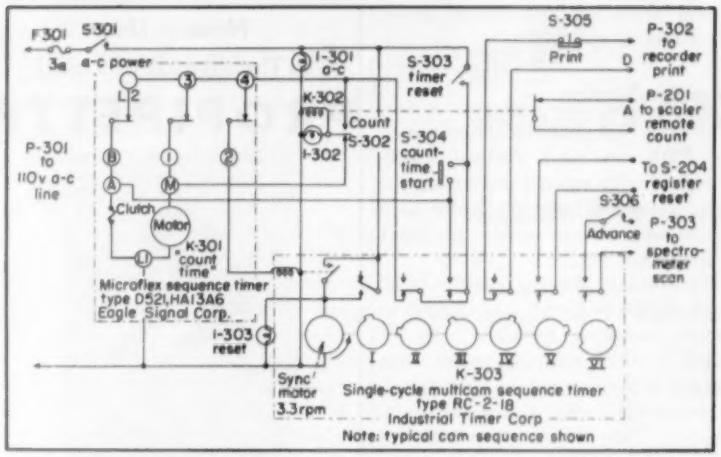

FIG. 3 Programing section of PDR. Mechanical connections of K-301 are not shown; operating sequence: timing—contacts L2-B, 3-1 closed, 4-2 open; time out—contacts L2-B, 4-2 closed, 3-1 open; reset—all contacts open

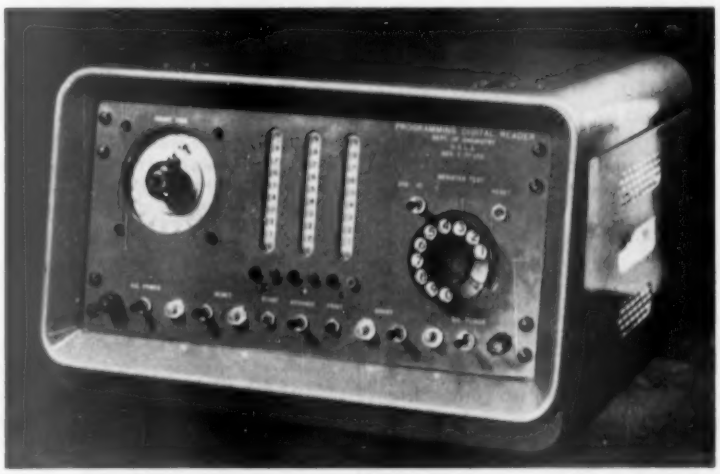

FIG. 4. Front view of programing digital reader

graming applications to replace the counting-rate meter or printing register and programing devices.

Applications

The PDR has been used for programming the readout in a magnetic beta-ray spectrometer (Fig. 1), in a single-channel pulse-height analyzer with a stepping switch to change the baseline voltage (*1*), and to program data recording for decay curves on a traffic counter using a variety of detectors, all with satisfactory results. The gamma spectrum of Na^{22} from a single-channel pulse-height analyzer has been recorded simultaneously by a traffic counter and the PDR with no deviations within the resolution of the recorder used.

For application where the logarithm of the count is desired (as in decay curves), the linear voltage divider could be replaced by a logarithmic one having the same applied potentials and total resistance, or a logarithmic recorder (*2*) could be used.

The basic instrument is also adaptable for use with conventional multichannel analyzers to reduce the time and labor involved in reading and plotting data. One register section similar to that described here would replace the original register in each channel, and the programing section would include a provision for reading each register in turn and plotting the counts accumulated *vs* channel number before resetting the register. The whole of the reading and plotting sequence could be completed in a few seconds with great advantage in work with short-lived nuclides. No special memory

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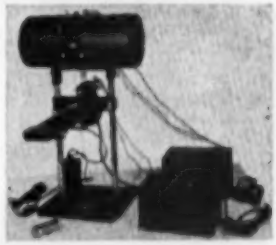

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circuit such as has been used previously (*3*) would be necessary.

Register Section

Used in conjunction with a scaler, the PDR acts as a decimal, 1,000-count-capacity register converting the register counts from the scaler to a decimalized voltage suitable for input to a recorder.

Three rotary stepping switches (*K*-1, *K*-10, *K*-100 in Fig. 2) are utilized as decade counters. Switches were selected in which stepping occurs when the coil is deenergized; i.e., the switch is electrically cocked and mechanically stepped. One of the contact banks on each switch is utilized to step once automatically to eliminate the eleventh, or "off," position for decimal switch operation (*1*). The high speed of self-operated stepping prevents any count loss due to the automatic step. The same bank of contacts on *K*-1 and *K*-10 is utilized to operate the next relay (*K*-10 and *K*-100, respectively) at the tenth count for decade counting.

At the end of counting, the reset relay *K*-202 can be operated manually by closing *S*-204 or automatically by the closing of contact *V* of *K*-303 (see Fig. 3). Relay *K*-202 then applies power to all decade coils through the interrupter springs and the normally-off switches. The decade switches step automatically to their zero position where a cam on each opens the normally-off switch, removing power from the coil.

A second bank of contacts on each relay (bottom bank in Fig. 2) operates neon register lamps (NE-2's) on the front panel (see Fig. 4), which gives a visual indication to the operator of the count registered on the switches. The lights provide a means of checking operation and standardization of the PDR. The recorder span is adjusted to coincide with the register of the PDR so the scale on the graph paper used reads register counts directly.

The third bank of contacts on each switch (center banks in Fig. 2) is connected to a precision voltage-divider network (*R*-2, *R*-12, and *R*-102) in such a manner that the output voltage at *J*-202 is directly proportional to the registered counts. Each of the switches has a separate network and current source so that the voltages registered on each can be added to give the proper output.

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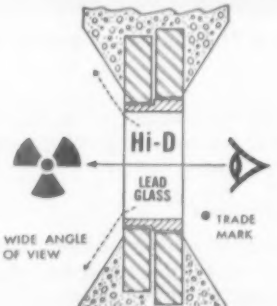

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through *J*-1, *J*-10, *J*-100 for standardizing the output voltage of each decade. Current adjustments may be made by means of screw-driver adjustments of *R*-3, *R*-13, and *R*-103.

For simplicity and stability the current sources for the networks are mercury cells (Mallory type RM-12R); *B*-1 and *B*-10 are single cells; and *B*-100 consists of nine cells in series. The tapped resistors *R*-2, *R*-12, *R*-102 have 10, 100 and 1,000 ohm steps, respectively, and with 1-ma current flowing in each, the output voltage steps are 0.0100, 0.1000 and 1.000 volt. Precision wire-wound resistors ($\pm 0.05\%$) have been used for the networks, giving the PDR an output-voltage accuracy well within the limitations of conventional recorders.

The recorder used with the PDR should have an input impedance greater than 100 kilohms to prevent loading of the voltage dividers, which would introduce errors in the recorded count. The 10-volt output total was selected for use with a Moseley Autograph model 1 recorder with printing pen.

To facilitate standardizing the recorder to the PDR and to facilitate checking register-relay operations, a telephone dial (*S*-203, "Register Test") and a three-position switch (*S*-202) are used to supply a given number of pulses to any of the decades. The recorder span may be adjusted so that the recorder reads correctly over the whole range using *S*-203 to set up any desired count on the register.

Programing Section

The programing section of the PDR controls two basic operations: timing of the counting interval and timing of a sequence of auxiliary operations occurring at the end of the counting interval.

Count timing is provided by a Microflex sequence timer, *K*-301, which can be set from the front panel to the desired counting time. At the end of this interval a single-cycle sequence timer, *K*-303, is energized. *K*-303 then operates a preset sequence of switches for automatic control of the PDR and other instruments.

The programing section can be used also for programing the operation of counting and recording instruments without using the register section with its recorder readout, if other types of data recording are desired in a particular situation. Thus, because of its wide range of programing possibilities,

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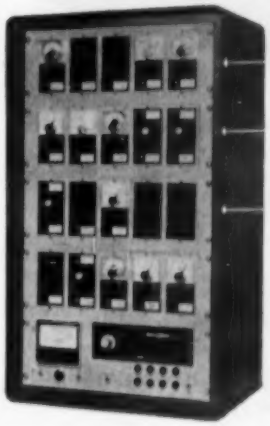

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the PDR can be used to replace several types of fixed-sequence programing devices commonly used in counting.

For automatic programed operation, the timer "reset" switch, *S*-303, is closed, and the count-time-"start" switch, *S*-304, is closed momentarily, supplying power to the motor and clutch of the "count-time" sequence timer, *K*-301. Contacts *L2-B* and 3-1 of *K*-301 then close, maintaining a-c power to the motor and clutch, in addition to starting scaler counting by opening relay *K*-302, which acts as the scaler remote-count switch. At the end of a preset time, set by the front-panel dial of *K*-301, contact 3-1 opens, stopping the motor of *K*-301 and opening the remote-count relay, which stops the counting. At the same time, contact 4-2 of *K*-301 closes, energizing the starting relay of the auxiliary sequence timer, *K*-303. This timer consists of a synchronous motor driving six adjustable cams, which in turn operate microswitches for auxiliary programing. The cams are adjustable both as to time open or closed and as to position in cycle of opening or closing, so that programing can be varied to suit the particular situation.

The cam sequence shown in Fig. 3 is a typical setting for beta-ray-spectrometer operation. As the motor of *K*-303 turns the cams, cam 1 allows its microswitch to close, maintaining power to the motor; cam 4 then opens and closes, causing the recorder to print the registered counts; cam 5 closes momentarily, resetting the register to zero; cam 6 closes, applying power to the spectrometer scan motor advancing the magnetic field to the next point; cam 2 then closes, resetting the Microflex timer *K*-301; cam 3 then starts *K*-301 so counting begins again; cam 1 opens, stopping *K*-303 ready for the next cycle. The reset light, *I*-303, shows when the motor of *K*-303 is running.

The counting cycle may be stopped at any time by the reset switch *S*-303. Provision is made so that *K*-303 always returns to the beginning of the programing cycle, even though *S*-303 is open. *S*-306, "advance," provides a means of opening the advance circuit to prevent advancing of the spectrometer if desired. A manual "print" button, *S*-305, is provided for use when automatic operation is not desired, and as an aid to adjustment of the recorder span to the PDR output. A "count" switch, *S*-302, is also provided for manual or automatic operation of the scaler count circuit. The "count"

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light, *I*-302, shows the scaler count circuit to be energized.

By suitable connection of either the normally open or closed contacts of *K*-303 and adjustment of the cam positions, any desired sequence of operation may be achieved.

* * *

This paper is based on part of a dissertation presented by D. S. Harmer in partial fulfillment of the requirements for the degree of Doctor of Philosophy.

This work was supported by the U. S. Atomic Energy Commission. The authors wish to thank Mr. K. K. Jensen and Mr. R. W. Farmer of the Electronics Shop of the Department of Chemistry for their assistance.

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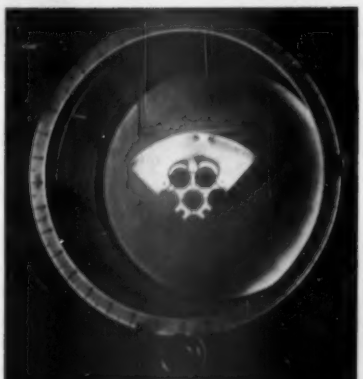

PHOTOELASTICITY REVEALS PRESSURE-VESSEL STRAINS. Westinghouse engineers are using photoelasticity techniques for design studies of reactor pressure vessels. Where numerous penetrations make it impossible to compute the strains that result from pressurizing, a plastic scale model can be used to display them. Engineers make a model of a special epoxy resin (see Figure at top for PWR model), heat it to a critical temperature, stress it with pressure, and slowly cool it under the stress. The result is a frozen-in stress pattern that becomes visible when the model or sections of it are examined between crossed polarizers. Photo at bottom shows stresses frozen into a section of a model

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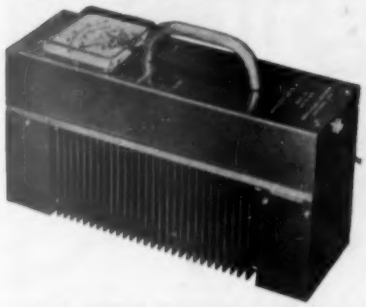

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Fusion Power— The Pace Quickens

The veil of secrecy that surrounds efforts at utilizing fusion reactions for power parted slightly when the U. S. Atomic Energy Commission announced that a "large experimental device for research into controlled thermonuclear reactions" would be built at Princeton University, site of the Project Matterhorn branch of the over-all Project Sherwood for controlled thermonuclear power. While no details were given on the Model C Stellerator (as the new device will be named) Commission spokesmen used the occasion to provide a retrospective look at some of the early fusion-power research at another Project Sherwood site, Los Alamos.

The Stellerator. Here's what was told about the new Princeton device: It is a research tool, very much larger than any Sherwood project machine built heretofore. Cost will be "a very substantial figure" . . . "a couple of scores of millions."

As implied by the model number this is not the first machine of the type to be worked on at Princeton.

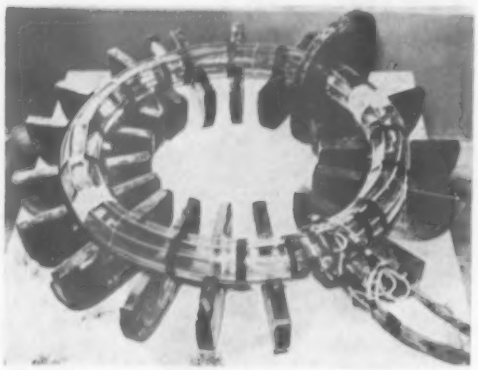
Construction timetable is as follows: architect-engineering work for supporting office and laboratory facilities will begin in the near future, construction in the summer. Cost will be $2.8-million. Facilities to house the device will get under way in 1958 and experiments with the new stellerator will not start till late 1960.

The stellerator is said to consist essentially of a hollow tube containing ionized gas. Around the tube are external coils which produce a magnetic field to confine the gas. The object—"to heat the gas to temperatures of millions of degrees and at the same time to confine the heated gas within the tube for sufficient time to allow fusion reactions to take place."

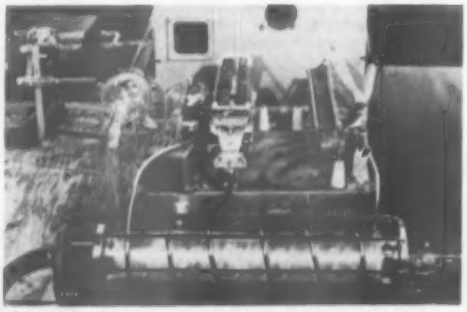
Nothing has been revealed about the size of the new stellerator, but the three-year construction period plus the fact that the Princeton researchers have outstripped their new $500,000 laboratory building would seem to imply a diameter of the order of over a hundred feet.

Early fusion research. AEC also revealed some details of early Sherwood devices at Los Alamos, devices whose very names were news when first revealed in NUCLEONICS (NU, Dec. '55, p. 23). These devices were used to study the pinch effect in which the self-magnetic field of a current causes it to contract into a tight pencil removed from the walls of the container.

The devices (see figures) fall into two classes: Perhapsatrons utilizing a toroidal electrodeless discharge similar to that studied by nonsecret groups at Tufts College and the University of Southern California (NU, Feb. '56, p. 43) and Columbuses using a linear discharge with end electrodes, which bear a striking resemblance to the types discussed by Russian scientist Igor Kurchatov in his now-famous Harwell speech (NU, June '56, p. 36).



DISCHARGE TUBE for Perhapsatron showing primary windings and iron cores of magnets used to heat and contain gas. Typically 4-μf, 15-kv condenser is discharged through turns that parallel doughnut-shaped discharge tube. Gas in tube is preionized with 27-Mc electromagnetic radiation



COLUMBUS-type quartz discharge tube with accompanying electrical circuits (1955)

PERHAPSATRON, Los Alamos' first (1953) apparatus for study of the pinch effect. Large structure is magnet used to carry flux through the center of doughnut-shaped discharge tube, shown at center

It's Time to Look at Nuclear Merchant Ship Safety

By R. A. FAYRAM* and H. J. SCHNEIDER†
Aerojet-General Nucleonics, San Ramon, California

Programs aimed at development and construction of nuclear-powered merchant ships are known to be under way or imminent in several countries, including the U. S., Norway, Great Britain, Sweden, West Germany and Japan. During the early development work, and certainly before any possible large-scale production programs, much thought must be given to safety measures for such vessels. "Third-party" responsibility will some day be imposed on ship operators, certain governmental agencies, labor groups, public health organizations, insurance firms, and possibly even the shipbuilders. The concern now is with developing an adequate set of principles and procedures for minimizing nuclear-ship hazards.

The central question of nuclear-ship safety concerns unforeseen accidents, such as ship collisions and sinkings, with the accompanying possible release of accumulated radioactive materials. The mobility of the ship and the medium in which it moves make this safety question different from the problem for land-based reactors.

Possibility of Accidents

The first consideration requires that a nuclear ship be designed with the anticipation that it may quite likely suffer an accident. With the present trend (*1*) toward supertankers and giant bulk-cargo carriers of up to 85,000 tons dead weight, it seems conservative to estimate from historical data that there may be about 7 accidents per year of minor to major degree involving ships of over 20,000 tons dead weight (the size of proposed nuclear vessels). Perhaps about 3 or 4 of these will occur in or near ports, or in contained bodies of water such as the Mediterranean, San Francisco Bay, etc. When nuclear-ship programs are in full stride, say about 1970, they may suffer one or two of these accidents each year near populated areas. Thus, although land-based reactors may be designed with the understanding that major catastrophes (earthquakes or floods) are quite unlikely, ship reactor designs must consider the fact that earthquake-proportion damage is a real possibility.

Water Contamination

The fact that the failure of a nautical reactor will occur over a body of water further complicates the hazard picture. Depending on the circumstances, the water could either suppress or accelerate the release of fission products to the atmosphere.

In addition, the amount of radioactivity present in a ship-size reactor is large enough to make contamination of the water body itself a consideration. A typical merchant-ship reactor providing a shaft power output of about 20,000 hp will have a heat output of the order of 60 Mw. After a long period of operation, the accumulation of fission-product activity within the reactor would reach a saturation value of ~6 curies per watt of power or a total activity of $\sim 3.6 \times 10^8$ curies. The table shows the contamination level after 10 days in various bodies of water with the addition of this much activity, assuming complete mixing. One sees that the contamination exceeds the maximum permissible concentration of activity for drinking water for many of the water bodies considered.

The assumption of complete mixing, however, is unrealistic except for very small water volumes. Actually the fission products will be disseminated from the point of release by diffusion and marine currents at a moderate rate. A rough idea of the distribution of contamination in a sizeable water body can be obtained by solving the diffusion equation for a point source in an infinite medium. If the dispersed fission products are assumed to behave like an ionic solution, a representative value can be assigned for the diffusion coefficient and the activity concentration can be calculated as a function of distance from the release point for various times. The figure shows the results of such a calculation for the 60-Mw reactor previously described. The specific activity is found to be lower than the tolerance level of 10^{-10} curies/liter for distances greater than a mile. Thus, as far as hazard to inhabited areas is concerned, water contamination will be important only when the ship is very close.

When the ship is at sea, the most urgent considerations will be exposure of the ship's crew and passengers and salvage operations. When the ship is in port, the shore-line contamination must also be evaluated and will involve contamination of the water as well as

Contamination of Various Bodies of Water*

Body of water	*Volume (10^{17} liters)*	*Contamination (10^{-10} curies/liter)*
Carribean Sea	9.573	0.25
San Francisco Bay	0.000076	22,400
Mediterranean plus Black Sea	4.238	0.55
Baltic Sea	0.023	102
Red Sea	0.215	11
Persian Gulf	0.006	392
Atlantic Ocean	323	0.0073
Pacific Ocean	707	0.0033
Maximum Permissible Concentration for drinking water		1

* For complete mixing of the saturated fission products from a 60-Mw reactor 10 days after shutdown.

* Deceased. Correspondence may be directed to Richard Brooks, Aerojet-General Nucleonics.
† Student, Department of Mechanical Engineering, University of California, Berkeley.

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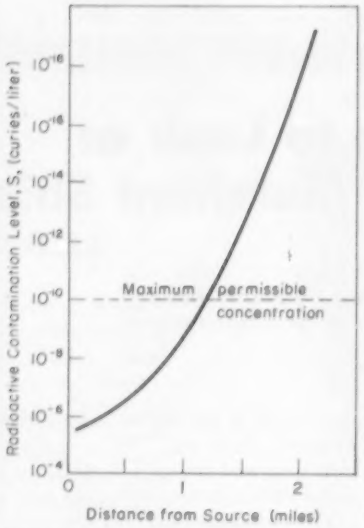

DIFFUSION in a large body of water of the saturated fission product activity from a 60-Mw reactor 50 days after a failure. 40% of the available fission products are assumed to be released from a point source

the usual atmospheric pollution. Compared with a land-based nuclear plant accident, the shore-line accident therefore presents the additional complications of salvage difficulties and transport of radioactivity by water.

Safety Measures

Since all possible precautions must be taken to guard against release of fission products, it is pertinent to consider some of the causes of failure and to suggest some methods of their prevention or minimization of their effects.

Fission products could be set free by an explosive release of nuclear energy. Fortunately, a well-designed reactor is generally a very sluggish machine in this respect; the maximum energy release from the nuclear explosion of such a reactor might be on the order of magnitude of a mild chemical explosion. An explosion of this sort might happen if the reactor is allowed to run away. To prevent such a runaway from being too serious, the reactor must be designed with strongly negative temperature and pressure coefficients of reactivity, and must provide fully dependable safety mechanisms for shutdown. Much attention (*2*, *3*) has been given this problem, so that further discussion is unnecessary.

Fission products could also be released by mechanical rupture of primary vessels and piping, with subsequent loss of coolant and mechanical, thermal, or chemical expulsion of hazardous materials. From a safety

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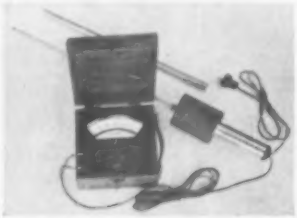

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viewpoint, it is evident that nuclear ship-propulsion systems using low-pressure coolant, even at high temperatures, are especially desirable. But this consideration must in practice be balanced against the improved thermodynamic-cycle efficiencies and reduced operating costs of high-pressure systems.

The containment of fission products and fuel materials might proceed as for land-based, stationary reactors. The power plant can be entirely enclosed in one or more suitable casings that will resist any conceivable explosion in the system, thus preventing the escape of any hazardous materials. This is a common approach to assuring the safety of nuclear power plants. There are two major objections to this scheme for nuclear-ship plants: first, the difficulty of maintaining important components such as pumps, valves, instruments, and heat exchangers, and second, the complexities of salvage of such nuclear plants in the event of a major catastrophe. Space limitations also impose some difficulty in the design of a containing vessel.

A possible solution may lie in minimizing the potential hazard of any single ingredient. The use of fail-safe detection and control mechanisms is, of course, implicit. Consideration of the use of homogeneous reactors (incidentally including but certainly not confined to the aqueous types) may prove fruitful. The possibility exists for this reactor type that the fission products can be continuously removed and stored so as to be completely safe. In this way the amount of activity available for release in the event of an accident would be substantially reduced.

Although maintenance, fuel changes, and salvage procedures also have a bearing, safety will probably have an important influence on the location of a reactor aboard ship. Historical data are not yet available on the probability and extent of accidents to engine-room spaces. Such data should be collected and reviewed with regard to the prospect of placing reactors in the normal locations aboard ship, or in unconventional positions.

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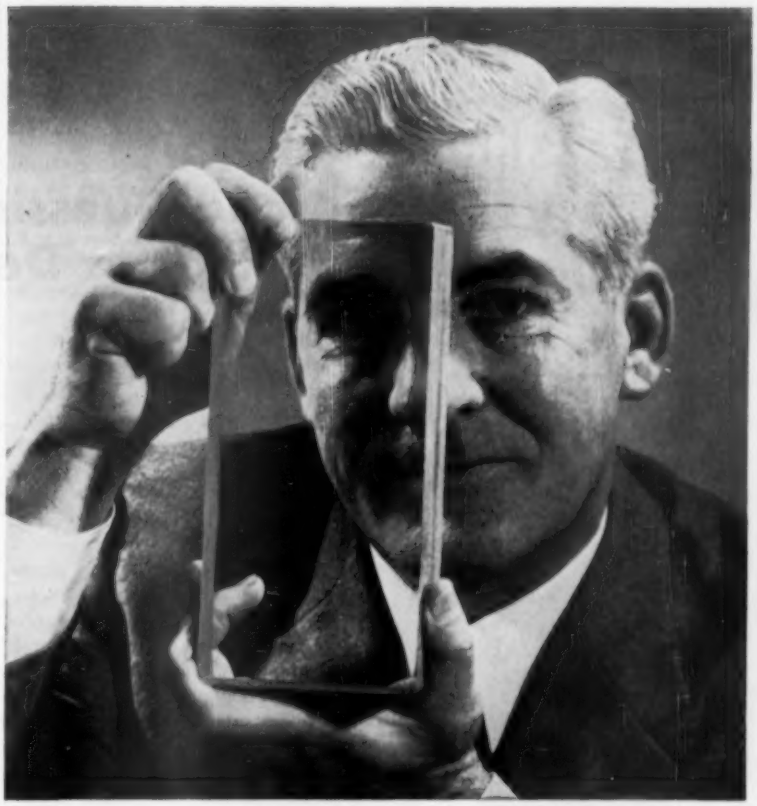

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LITR Experience

Second-Guessing the Reactor Designers

By J. A. COX
Oak Ridge National Laboratory
Oak Ridge, Tenn.

In performing his day-to-day duties the reactor operator is bound to acquire information and experience concerning his particular facility that was not available to the original designers. He will probably always become aware of specific problems of routine operation that could easily have been eliminated by the reactor designer if he had foreseen them. The reactor operator's comments then should be of interest to groups now engaged in reactor design. Accordingly some comments based on the operating experience with the Low Intensity Test Reactor (LITR) at Oak Ridge National Laboratory are presented here in the hope that they will be of assistance to reactor designers.

The LITR was constructed primarily as a mock-up for hydraulic and nuclear testing of the design of the Materials Testing Reactor (MTR). LITR is cooled by circulating ordinary water and has fuel elements containing aluminum-clad plates of enriched uranium-aluminum alloy. Upon completion of the design testing the reactor was shielded and fueled for operation at about 1 Mw for use in the research and development programs at ORNL. It was the first of the high-flux research reactors.

Lessons Learned

Some of the important lessons that experience has provided follow:

Accessibility from above. The reactor should have maximum accessibility from above. Many experiments can be inserted in vertical tubes that extend into core spaces in the lattice. These tubes carry thermocouple leads, cooling and pressure lines, etc. to apparatus outside the reactor. Any structure or machinery that obstructs access from above the lattice makes insertion and removal of experiments more difficult. In LITR and MTR the rod-drive assembly which is suspended below the top plug must be completely removed to work without interference when inserting and removing experimental apparatus. The Oak Ridge Research Reactor (ORR) design eliminates practically all such interference above the lattice by having the shim rods driven from below.

Beam holes. Beam holes should have sufficient clearance between plug and liner to minimize galling and sticking. In the LITR, experience has shown that at least $\frac{1}{8}$-in. clearance should be provided in a 6-in.-dia. beam hole. Current practice is to leave about $\frac{3}{16}$ in. In smaller holes, such as those used for pneumatic tubes, severe galling has been experienced where only $\frac{1}{32}$-in. clearance was left in a $1\frac{1}{4}$-in. hole. In addition selection of materials is important; for example, aluminum tubes in aluminum liners are apt to gall. It is preferable to make one of the surfaces of stainless steel. Shielding doors in front of each beam hole have been found to be a great convenience.

Radioactive gas disposal. Radioactive gas is encountered in the LITR water system when the reactor tank is opened. Even though an off-gas suction is provided on a surge tank, considerable radioactive gas accumulates in the water. Provision should be made in the design for an off-gas arrangement to remove the gas.

Water purification. The longer-lived radioactivities in the cooling water are formed from activation of impurities

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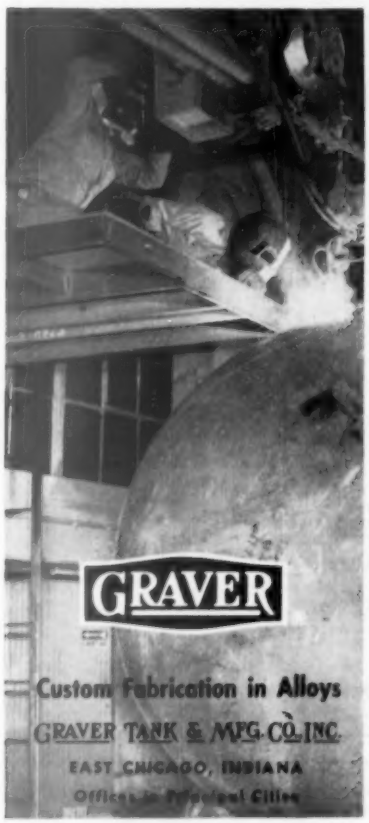

and from such reactions with materials in the lattice as AL^{27} $(n,\alpha)Na^{24}$. It is desirable to purify the water continuously to remove these radioisotopes so that tools and other objects immersed in the reactor tank water are not badly contaminated. Purification of the water also reduces corrosion of reactor components. In the LITR contamination is continuously removed from the circulating water with a bypass demineralizer consisting of cation, anion, and mixed-bed resin columns. The demineralizer enables the *p*H to be kept between 5.5 and 6.5 and the resistance of the water maintained above 1 megohm-cm. This concentrates the radioactivity on the resin columns, which must be shielded. Filters in the water system also become radioactive and should be placed in a remote location or shielded.

Component removal. In the LITR, where fuel elements often last for two years, some trouble has been experienced with elements sticking in the lower grid. It is believed that close-fitting endboxes should be avoided to minimize this problem.

In general, reactors should be designed so that components can be replaced easily by remote methods where necessary. Some components can remain in place for ten years or more before removal. The long-term effects of corrosion and radiation on ease of removal of various parts should be considered in the design.

Sealed canal linings. Pools and canals made for use with a reactor should be lined with impervious material. If bare concrete is used, radioactivity in the water can soak into the walls giving a source of radiation that is very difficult to eliminate. Paint can be satisfactory, provided fuel elements are not placed so close to a painted surface as to damage the paint by radiation and provided that it is possible to drain and repaint when necessary. Tile is satisfactory if the mortar joints are treated to minimize the absorption of radioactivity. Other materials considered for this use are aluminum and stainless steel.

Canal decontamination. Provisions should also be made to repurify canal water continuously to remove radioactivity and to reduce corrosion of stored materials. Since silt settles, it is desirable to have a system for pumping the silt off the floor to a filter or sump. This silt can be radioactive and provision should be made for this problem in the filter or sump drain.

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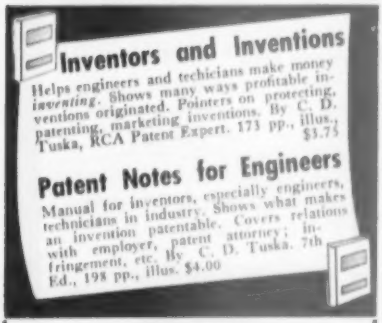
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How Radioactive Wastes Will Be Handled at PWR

By J. R. LaPOINTE
Bettis Plant
Westinghouse Electric Corp.
Pittsburgh, Pennsylvania

When the Pressurized Water Reactor (PWR) is completed at Shippingport this year, it will be a source of radioactive wastes as well as of electric power. The sources of wastes in PWR are varied: in effluent coolant water they can arise from activation of corrosion products and fission-product release due to possible fuel-element rupture; tools that have become activated may need to be discarded; spent resin used to purify coolant water will be active after use; etc. The quantities and types of wastes are given in the table.

Design Criteria

The radioactive waste disposal system at PWR has been designed to meet stringent limits on the concentrations of pollutants released to the environment. The radioactive-waste concentration in the water discharged to the Ohio River, and in the air at the point of discharge to the surrounding atmosphere, will not exceed 10% of the maximum permissible concentrations (*1*) recommended in Table 5 of National Bureau of Standards Handbook 61 (*2*).

Because data on the quantity and activity of all the *specific* radioisotopes to be encountered in the PWR coolant were not available during the design of the waste-disposal system, the design criterion for disposal of nonvolatile material was that the concentration should not exceed 10% of the value listed* for an *unidentified* mixture of radioisotopes. In general, this limit is more restrictive by a factor of 100 than the corresponding limits for specific isotopes.

Another design criterion is that sufficient capacity must be provided to store radioactive gases for the maximum expected duration of a weather inversion—5 days, based on the Donora incident (*3*).

Waste-Disposal System

To meet the requirements just outlined, the PWR waste-disposal system uses (a) natural decay; (b) concentration and storage; and (c) dilution. These methods are applied, in different ways for the various categories of radioactive wastes generated in PWR, but all treatment is carried out in a central waste-processing area. Here the wastes are segregated, processed or

* Standards for protection against radiation, 10 CFR Part 20 (U. S. Atomic Energy Commission, Washington, D. C., 1957).

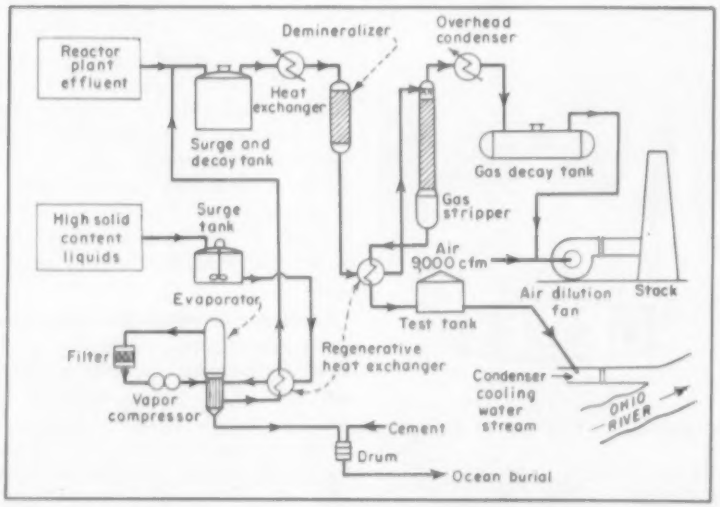

FIG. 1. Flow diagram of radioactive liquid and gaseous waste disposal system. Activity discharge to Ohio River will not exceed 0.03 mc/day

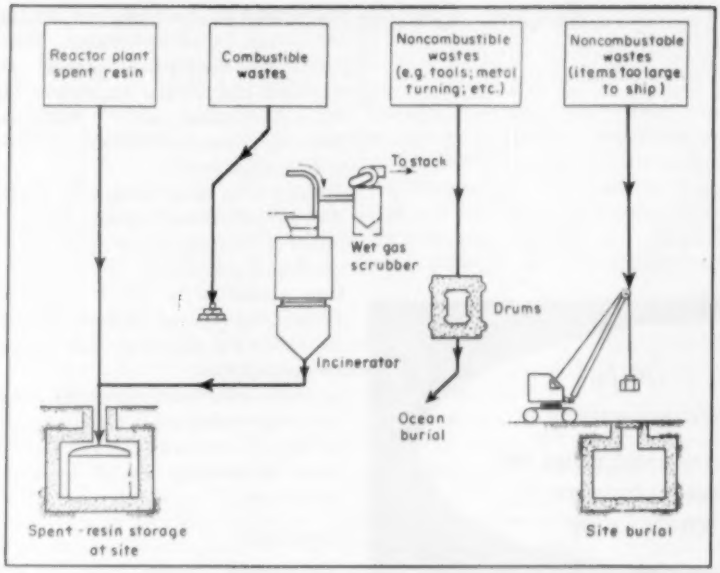

FIG. 2. Flow diagram of radioactive solid waste disposal system. Combustible wastes are burned in incinerator. Gases of combustion are scrubbed and filtered and ashes and wash water transferred to storage tank

stored, and ultimately disposed of as outlined in the following paragraphs.

Liquid and gaseous wastes. Reactor plant liquid effluent (see Fig. 1) is the major source of radioactive waste. It is delivered to underground stainless-steel tanks where its radioactivity is measured. Depending upon the results of the measurement, the liquid waste may be stored in these tanks while the radioactivity decays and/or processed through demineralizers and a gas stripper, or, if its radioactivity is within permissible limits, the liquid can be blended with the 58,000-gpm condenser cooling stream for discharge to the Ohio River.

Soluble radioactive impurities as well as particulate matter are removed in a series of mixed-bed demineralizers. Dissolved fission gases, if present in these wastes, are then removed in a steam stripper operating under slight vacuum, and are stored in steel tanks until sufficiently decayed (10^{-7} μc/ml) to permit safe discharge to the atmosphere. The processed liquid is then held up in test tanks for sampling to ensure that the liquid discharged from the plant will meet allowable tolerances (10^{-8} μc/ml) when diluted with the condenser cooling water stream.

Liquid and gaseous effluents will be monitored in the control area both before and after discharge of all wastes to the environment and at a sufficient number of sampling stations outside the plant area to ensure safe practices.

Liquid wastes containing a high concentration of impurities, such as fluids used for decontaminating equipment, are processed through a vapor-compressor evaporator. The distillate from the evaporator is sent to the liquid surge tanks, and the concentrate is mixed with cement and put in drums for burial at sea.

Solid wastes. Spent demineralizer resin (see Fig. 2), principally from the internal purification system, is diverted to permanent underground stainless-steel storage tanks surrounded by a waterproof concrete enclosure. In the tanks, solids settle out and the water used to transport the resin is transferred to the liquid-decay tanks for possible further processing.

Combustible contaminated solid

Radioactive Wastes at PWR

Type	*Monthly amount*	*Specific activity* (μc/cm³)* Volatile	Nonvolatile
Reactor plant effluent	< 16,000 gal	7.6	1.7
High-solid-content liquids	7,500 gal		< 0.27
Spent resin	36 ft³		10^4
Combustible wastes	3,000 lb (avg) 10,000 lb (max)		Variable

* Total activity of liquids and gases: ~20 c/day

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wastes such as paper, rags and clothing are burned in an incinerator. Gases from the combustion process are scrubbed and filtered to remove airborne particulate matter. Ashes and wash water are then transferred to the resin storage tank.

Small solid noncombustible wastes, such as tools, metal turnings, and small pieces of equipment, are placed in 30-gal drums and sealed. Each drum is then placed in a larger drum and shielded by pouring concrete between the two. The drums are then shipped for ocean burial.

Larger solid noncombustible wastes too large to ship are handled remotely and stored in an underground concrete vault or suitably buried in a site "graveyard."

Conclusion

Experimental data developed since the initial design of the plant indicates that the radioactivity of the waste will be considerably less than assumed here. Further conservatism is provided in that the system was designed as a batch process discharging periodically to a continuously flowing pure condenser cooling water stream. The only source of waste liquid to the river from the plant will be the addition to this circulating stream. Operation of the plant will prove the degree of conservatism and will provide data that will establish modified criteria for the design of radioactive waste disposal facilities for future nuclear power plants.

A monitoring program was started early in 1956 to determine the types and variation in the amounts of radioactive materials which occur naturally in the environment around the PWR site. The program will extend over a period of ~1½ years prior to startup so that similar studies conducted after startup will ensure that the system meets allowable tolerances.

* * *

Based on a paper presented at the 2nd Inservice Training Course on Radioactive Liquid Wastes at the University of Michigan, November 2, 1956.

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NUCLEAR ENGINEERING NOTES

Highlights from Nuclear Congress

The emphasis was on fuel cycling in power reactors at the 2nd Nuclear Engineering and Science Conference held recently in Philadelphia. A full six sessions were devoted to discussions of processes for plutonium separation and fuel decontamination, plans for remote refabrication of reclaimed fuel, and theories of the equilibrium-buildup concentrations of heavy isotopes in recycled fuel. In addition there were reports on the physics, the engineering, and the economics of various AEC reactor projects.

Fuel-cycle theory. Manson Benedict (M.I.T.) explained the neutron balance equations governing the changes in fuel composition and reactivity during a cycle of a single-region thermal reactor. He then derived the limitation on fuel burnup due to reactivity loss from changes in fuel composition.

In applying the method to a typical thermal uranium-plutonium power-reactor design Benedict considered four different modes of operation—batch irradiation of the fuel with and without plutonium recyling and continuous movement of the fuel through the core during the irradiation with and without plutonium recycling. By assuming reasonable costs for the fuel materials and the various fuel reprocessing operations he arrives at the total fuel cost per unit of energy produced for each of the four cycles. For all the cycles fuel costs rise sharply for burnup fractions less than 1% (or 10,000 Mwd/ton). Continuous irradiation turns out to be more effective than Pu recycling in reducing the total fuel cost.

Remote fuel fabrication. A. B. Shuck (Argonne) outlined the problems of fabricating solid-fuel elements from radioactive fuel material by remote operations. This task arises for Pu and U^{233}, which are naturally active, and also for reclaimed U^{235} that still contains fission products. ("Dirty uranium" of this kind can be used in fast reactors.)

Shuck presented the conceptual design of the remote mechanism being developed by Argonne to refabricate fuel for EBR-II. The device manufactures cylindrical fuel slugs from reprocessed fuel material that can have an activity as high as 10 c/gm. Molten fuel is injected under pressure into a precision ceramic mold. Dry runs have produced 2-ft-long, 0.140-in.-dia fuel cylinders with a diametral tolerance of 0.001 in.

Stress corrosion cracking. James A. Lane reported for E. G. Bohlman and G. M. Adamson (Oak Ridge) on stress-corrosion-cracking problems in the Homogeneous Reactor Test. Cracks perpendicular to lines of stress and extending up to 0.030 in. in depth were discovered in the secondary water system used to detect leaks at flange joints. The stress in some of the affected areas was as low as 5,000 psi. Chlorine in the secondary system is believed to have caused the cracks via a corrosion mechanism. Although this phenomenon has been observed before, the exact role played by the chlorine in producing the cracks is still not understood. Experimental evidence indicates that to avoid this problem the chlorine content of the water should be kept below 5 ppm. Lane pointed out that the problem is not peculiar to the HRT but would be present in any stressed austenetic-steel system containing water.

Half-fast limitations. In reviewing the highlights in reactor theory presented at the winter meeting of the American Nuclear Society in Washington, Alvin M. Weinberg, director of ORNL, discussed the fast-thermal reactor concept now being pursued by Argonne (NU, Jan. '57, p. 94). Weinberg suggested that the reactor could run into difficulty because it may not be as insensitive to fission-product poisons as an ordinary fast reactor. The problem arises from a generally unappreciated fact—the absorption cross section of fission products for neutrons in the region below 100 kev is quite high. At 25 kev Weinberg estimates that the fission-product absorption cross section is one-third the U^{235} fission cross section.

Conventional costs count. In discussing the engineering and construction of the Engineering Test Reactor in Idaho, P. D. Bush (Kaiser Engineers) made the point that the cost of the reactor itself (the pressure shell and everything inside it) amounted to only 20% of the total cost of the facility. Most of the money went into the reactor buildings, primary and secondary coolant systems, and supporting facilities. Bush observed that efforts directed toward cutting costs in the reactor core are therefore unrealistic unless similar efforts are applied to the other components of the reactor facility.

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Hydrogen Capacity of Metal Hydrides

Studies at Battelle Memorial Institute have been directed towards increasing the hydrogen content of metal hydrides. They reveal that alloy hydrides do not differ markedly in hydrogen capacity from the pure hydride over a wide temperature range. On the other hand metallic compounds show a significant decrease.

The significance of metal hydrides lies in the fact that they have a H content comparable with that of water combined with a temperature capability as high as 2,000° F—particularly the hydrides of vanadium. Thus hydrides of vanadium, niobium, titanium, and especially zirconium are being investigated as possible reactor materials. (See article by McCullough & Kopelman, NU, Nov. '56, p. 146.)

Trzeciak, Dilthey and Mallett of Battelle have investigated the effect of composition and preparation on several transition metal hydrides in an effort to improve the hydrogen content and thermal stability of these materials.

By using higher purity materials they were able to produce vanadium and niobium hydrides that had greater H content than ever reported. They conclude that the theoretical hydrogen limits are probably VH and NbH and that these might be obtained by using very pure materials.

The effect of the addition of alloy metals of various concentrations on the maximum hydrogen capacity of titanium and zirconium hydrides was also studied. With titanium they alloyed V, Nb, Mo, Mn, Cr, and Zr in various amounts. The highest H content was found for Ti alloyed with 69% V. This hydride showed a 22% increase in H concentration over pure Ti hydride. The best Zr alloy proved to be one containing 2.3% Ni, which had 20% more H than pure Zr hydride. Atomic size does not have a significant effect on H capacity.

The temperature dependence of hydrogen retention in various Zr alloys was studied. At 1,600° F the biggest effect was a 20% increase observed in a 2.3% tantalum alloy. At 2,200° F all the alloys were slightly worse than pure Zr.

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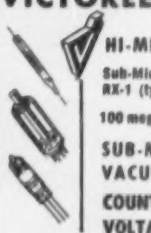

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Radioactive Inks Make Better Copies

Low-energy beta emitters distributed with the pigments in drawing inks produce a drawing that reproduces itself with exposure to photographic paper. The reproductions that are made qualify as "fine prints" among artists and museums. This is because the penumbra effect that makes an ordinary photocopy of small artistic value is missing in the "electron-printing" process. For example, the gray tone that replaces dead white areas in a photocopy does not appear in the electron print.

The process is compared by its inventors to stone lithography and copper etching. The radioactive original is comparable to the stone or plate, and the copies are "originals" in the sense that lithographs or etchings are.

The method has been invented and patented at the Louisiana State University by Harry E. Wheeler, Mrs. Naomi Wheeler, and Mrs. Caroline Durieux.

Low-energy betas combine safety and precision. Safety results from their low penetration, and precision of reproduction is a consequence of their short range. Five isotopes have been found useful: S^{35} (167 kev, 87 days), C^{14} (158 kev, 5,600 years), Ni^{63} (63 kev, 80 years), Pm^{147} (230 kev, 2.6 years), and Ca^{45} (250 kev, 160 days). Inks are generally made in concentrations of $\sim$1 mc/cm^3, and at this strength exposures of 1–8 days are required for reproduction.

Radiation Produces New-Type Plastics

New and better plastics can be made by radiation-induced copolymerization. The term is used to describe a process in which a monomer is grafted into a polymer of a different type. In *graft* copolymerization radiation creates a radical at a point on the polymer chain. Molecules of the monomer grow in a chain-reaction fashion from the site of the radical. The eventual structure is described as feather-like, with the original polymer chain forming the backbone or quill and the monomer chains forming the barbs that run out from the center. In *block* copolymers the original chain is fractured and monomer molecules are grafted into the break. The result is a single polymer chain with alternating sections of polymer B and monomer A. Depending on the situation created by the experimenter, radiation can induce graft copolymerization, block copolymerization, or both at once.

In many cases significant improvement of the original plastic occurs. Thus vinyl carbazole can be grafted to polyethylene to make a dielectric capable of withstanding higher temperatures than the original polyethylene. Increase in dielectric strength is accompanied by improvement in the tensile properties of the polyethylene. Styrene grafted to polyethylene and sulfonated makes a strong cation-exchange membrane of low resistivity and high permeaselectivity. Styrene can be surface grafted to Teflon for improved adhesive properties. When acrylonitrile is grafted to polydimethyl siloxane, the solvent resistance of the latter is improved.

Of several methods that have been used to produce grafting, most effective is to suspend the polymer film in a solution of monomer and irradiate the system. The dose used is generally $\sim 10^7$ erg/gm of Co^{60} gamma radiation. Monomer that is homopolymerized in the process—that is, made into polymer of its own type of molecules with no linking to the other type—can be washed away with solvent, and the change in weight indicates the amount of material successfully grafted. In most cases the graft is proportional to total dose, but in some cases there is autocatalysis, and the rate of grafting becomes greater as the process progresses. In general the process occurs most favorably when a radiation-susceptible polymer is irradiated in the presence of a monomer with a low susceptibility to homopolymerization.

Interesting results appear when a film is irradiated first and then immersed in monomer. Samples irradiated in vacuum experience more grafting than those irradiated in air. Unirradiated controls show no grafting. The effect is apparently due to free radicals, and paramagnetic-resonance experiments have shown that they remain effective for several months after the irradiation. Films continue to gain weight over this length of time.

Experiments directed toward more refined understanding of the process and of the parameters that govern it are being conducted under the direction of D. S. Ballantine of Brookhaven National Laboratory and R. B. Mesrobian of Continental

Can Co. (formerly at Polytechnic Institute of Brooklyn).*

Radiation Has Its Competitors, Too It's a competitive world, and radiation has competitors for markets that are growing as radiation applications are. It is too easy to compare new radiation uses with the techniques they might replace rather than with other techniques in the development stage that will bid for the same markets. For example:

Ultrasonic tracing. To find how fish behave around dams and spillways, fish-and-game people have developed a small capsule that contains an ultrasonic signal generator. It is only 2⅜ in. long and 1 in. in diameter, and it rides comfortably on the tail of a fish that is to be followed. Its battery-transistor-transducer system produces pulses that can be followed by an automatic tracking device mounted in a boat. The battery will keep the device operating for ~7 hr.

F-M-radio tracing. Zworykin and associates of RCA, working with the Rockefeller Institute, have made an f-m transmitter in an even smaller capsule. It is intended for studies in which the capsule passes along a human gastro-intestinal tract. Modulation by a diaphragm indicates pressures along the route of travel.

Linear polyethylene. Irradiated polyethylene is stable to temperatures up to ~300° F, so it can be heat sterilized. But linear polyethylene, made by new catalytic processes is cheaper and stable to ~240° F. It, too, can be heat sterilized.

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* W. K. W. Chen *et al.* Studies on graft copolymers derived by ionising radiation (Colloid Symposium, Madison, Wis., June, 1956). D. S. Ballantine *et al.* Studies on radiation-induced graft copolymerization and solid state polymerization (Progress report on fission product utilization. ix. Brookhaven National Laboratory, October, 1956). D. S. Ballantine, *Modern Plastics* (to be published).

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Isotope Milker Supplies Ba^{137} from Parent Cs^{137}

By R. L. NEWACHECK, L. J. BEAUFAIT, Jr, and E. E. ANDERSON
Tracerlab, Western Division, Richmond, California

Many radioisotope applications requiring repeated use of very short-lived activities have not been explored because of the difficulty of getting a continuous and reproducible supply of a suitable isotope where it is needed. Many have been impossible. The difficulty of scheduling shipment from a reactor or cyclotron has made uses of this type uneconomical and impractical.

The half-lives we are considering now are in the minutes range—short enough that the activity essentially disappears before there is hazard to operating personnel or contamination of the product involved. To provide a convenient source of short-lived isotopes we have developed a gamma milker* using ion-exchange separation and an automatic mechanical injection system to supply repeated amounts of a short-lived gamma emitter from a long-lived parent (Fig. 1).

Usable Isotopes

A large number of long-lived-parent–short-lived-daughter relationships exist in the more than a thousand radioisotopes now known. However, only a small number have practical significance from the standpoint of availability and appropriate decay characteristics. The most useful pairs are strontium-90–yttrium-90, barium-140–lanthanum-140, and cesium-137–barium-137. We have chosen the pair Cs^{137}-Ba^{137} because of the half-lives and radiation characteristics of the two isotopes. The daughter activity, Ba^{137}, has a moderately energetic gamma, 0.661 Mev, and a short half-life of 2.60 min. The parent, Cs^{137}, is a pure beta emitter (0.55 Mev) and has a long half-life, 30 years. The Ba^{137} can be readily separated from the Cs^{137} with ion-exchange resins and the proper rinsing solution (elutriant). For simplicity of use and handling in an automatic system, the ion-exchange-resin column containing the Cs^{137}-Ba^{137} activities and the elutriant solution reservoir have been especially designed in a single unit as a cartridge.

* Patent applied for.

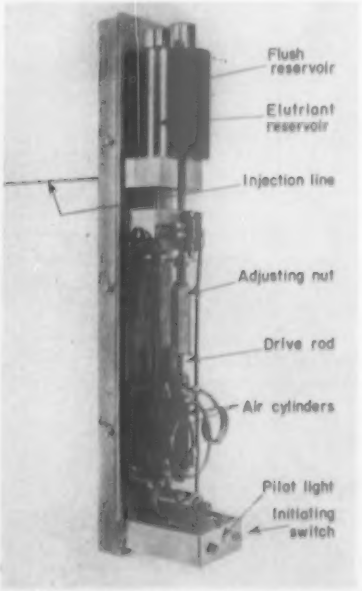

FIG. 1. Gamma milker with cover removed

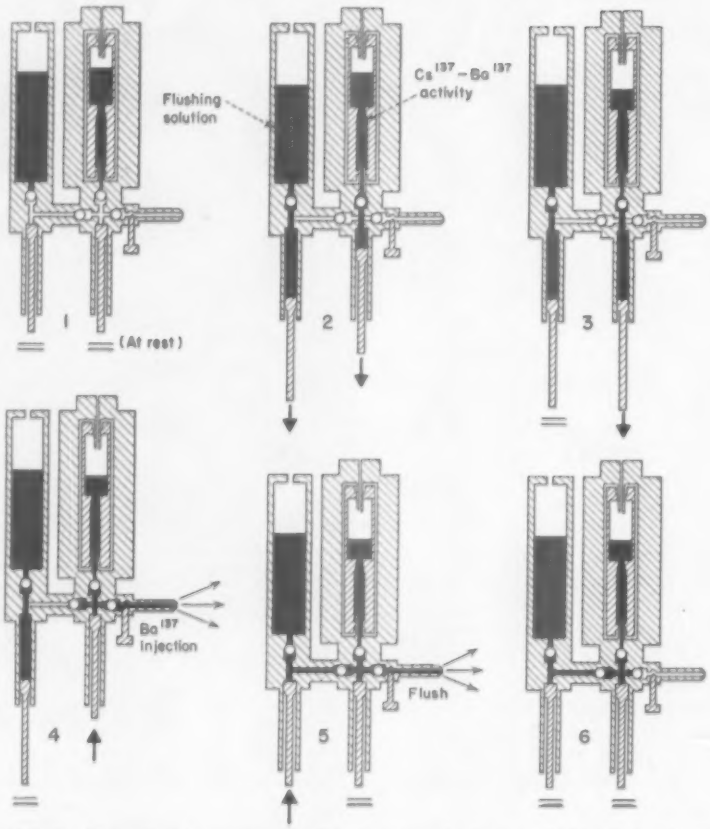

FIG. 2. Operational sequence of milker is automatically controlled by limit valves in pressure system, some of which can be seen in Fig. 1

Operation

For nearly all industrial applications injection must be made against relatively high pressures. In many applications the physical quantity of foreign material injected into the system must be kept to a minimum. This qualification has been a major factor in sizing the resin column, which in turn sets the physical quantity of the Ba^{137} solution injected. This value for the standard cartridge is 2.5 cm^3. Because this volume is relatively small, it is necessary to flush the in-

jection line completely to ensure that all Ba^{137} solution is injected. Figure 2 shows the six operational steps of the milker, which follow one another automatically. In step 1 the milker is at rest, with the cartridge in place. Step 2 takes place on an electrical pulse actuation. A measured amount of flushing solution is rapidly withdrawn into the flushing cavity, while the solution containing the Ba^{137} is beginning to be drawn slowly into the right-hand cylinder. Step 3 shows the flush cylinder at rest as the Ba^{137} solution continues to flow slowly into the right-hand cylinder. In step 4 the Ba^{137} is rapidly injected, followed instantaneously by the flush solution (as shown in step 5). After the injection, the milker is ready for the next injection (step 6 and step 1).

The system is initially loaded with enough elutriant and flushing solution to supply the number of injections for which it is designed ($\sim$20). Thus reloading is unnecessary; it is only necessary to supply vents at the top to admit air as the liquids are withdrawn.

The automatic sequencing of the milker is accomplished by a pneumatic system. The initiating air valve is operated by an electric solenoid. In addition, an *in operation* pilot light is electrically controlled. Electrical parts are hermetically sealed to eliminate open arcs. However, it is a simple matter to use an air pulse to initiate the operation and completely eliminate the need for electrical power.

Figure 1 shows the complete assembly with the cover housing removed. The two air cylinders which withdraw and inject the Ba^{137} solution and flush solution are immediately below the drive rods. The rods have adjusting nuts for changing the lengths of the strokes, which in turn determine the liquid volumes injected.

A manually operated stop valve on the side of the cylinder block, not shown in the figure, permits positive cutoff in the event of an emergency or when the injector is left on an operating pipeline for long periods without use.

The initiating switch and *in operation* pilot light are on the lower U-shaped bracket and are visible when the cover housing is in place.

By adjusting the length of the drive rod on the flush piston, it is possible to set the amount of flush solution to just clear the injection line of the radioactive solution. In addition, the rod for the piston that withdraws the Ba^{137} solution can be adjusted to vary the volume of radioactive solution—for example, to withdraw only 0.5 cm^3 from a cartridge designed for this volume. By using a very short injection line, the total injection—active and flush solutions—can be limited to 1.5 cm^3.

The air supply can be an instrument air line or a commercial compressed-gas cylinder. Since less than 0.1 scf (ft^3 at standard conditions) of gas is used for each injection, any small compressed-gas cylinder is sufficient for many operations. The injection pressure is a direct function of the pressure of the air supply. It is $\sim$8 times the gas pressure minus 100 psi. That is, for a 50-psi air supply, the milker will inject against a pressure of approximately $8 \times 50 - 100 = 300$ psi. The maximum injection pressure is limited to $\sim$600 psi by the injector.

Design Problems

The basic problem in the milker development is to produce, in a single easily handled unit, a cartridge giving the maximum number of injections with a minimum amount of Cs^{137} leakage injection.

The considerations involved can be separated into chemical and physical problems related to the selection of resin, elutriant solution, and construction materials. The problems are as follows:

1. Selection of the best ion-exchange resin—type, cross linkage, and particle size—to provide the greatest and fastest separation of the two radioisotopes
2. Selection of the best resin from the standpoint of radiation stability
3. Selection of the best elutriant solution and its concentration to give not only the most efficient separation but also the fastest release of the daughter activity
4. Determination of the effects of storage time (particularly radiation effects) on the cartridge after manufacture between routine successive injections and between random injections for short- and long-term periods
5. Determination of the millicurie capacity of the final cartridge

Cartridge

The investigation of these variables has led to the cartridge of Fig. 3. It consists of a 10-cm ion-exchange-resin column on which is adsorbed the Cs^{137} parent activity. The glass resin column is connected to a polyethylene

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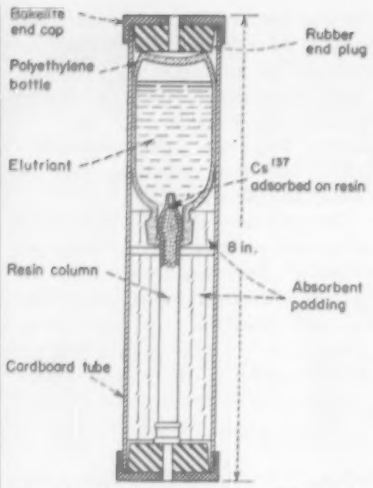

FIG. 3. Cartridge contains elutriant in polyethylene and 10-cm ion-exchange column

bottle containing the elutriant. Column and reservoir are loaded and sealed to eliminate all air and gas bubbles and ensure correct operation after shipment. The entire contents are rigidly packed in a reinforced cardboard tube filled with more than enough absorbent wadding to absorb completely the entire liquid contents if any accident should occur during shipment. Since the efficiency of separation with ion-exchange resins depends on complete resin-liquid contact at all times, the cartridge must remain in an upright position once it has been connected to the mechanical injection system and opened to the outside atmospheric pressure.

Connection to the mechanical injection system is made through the self-sealing rubber cap at the tip of the column by means of a syringe-type needle. The injector contains a cover for the upper end of the cartridge with a second syringe-type needle that vents the polyethylene bottle when the cartridge is installed.

After the appropriate number of injections of Ba^{137} have taken place, the spent cartridge is removed. In replacing the cartridge, the end caps of the new cartridge are removed and the new cartridge is inserted. The standard cartridge is designed to hold up to 5 mc of Cs^{137} and to supply twenty 5-mc injections of Ba^{137}, each in a volume of 2.5 cm^3.

Long-term storage effects have not yet been thoroughly investigated, but short-term effects are known. In Fig. 4 data are presented to show leakage vs number of shots. About 0.03% of the original Cs^{137} parent activity is contained in the twentieth injection. These data were taken with a cartridge containing 5 mc of Cs^{137} activity and after a storage time of 10 days from preparation. The data shown in this study were taken under the least favorable condition of rapidly repeated withdrawals. In normal operation, a minimum of 10 min would be allowed between injections to allow maximum growth of Ba^{137} daughter activity, and under these conditions we expect less Cs^{137} leakage than is shown in Fig. 4.

Industrial Uses

The potential uses of this milker in process control, routine testing, and experimental measurements are expected to be many. Especially in process control, in which the use of tracer isotopes is considered to be in its infancy, it is hard to estimate the extent of future applications.

In tests of fluid flow and leakage the device has several simple applications:

Leak detection. Leakage in a heat exchanger can be determined by making an injection at the inlet and measuring the radiation from the liquid leaving both the shell and tube sides of the exchanger. The ratio of the radiations of the two outlets will give a quantitative indication of the leakage.

Flow distribution. Flow studies can be made to determine the flow characteristics of large pipes or streams of liquids or fluidized particles. The injection is made upstream from the location to be studied. Detectors are located at various points throughout the stream, or samples are taken for counting.

Fluid-flow rates. One of the most useful applications of the milker is that of measuring flow velocities of liquids, or fluidized solids. Figure 5 shows the arrangement of the equipment for a typical flow measurement.

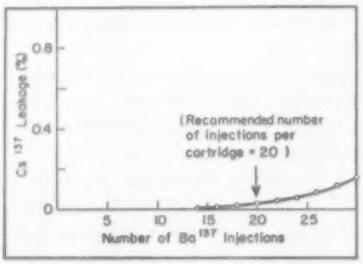

FIG. 4. Under worst conditions leakage of original Cs^{137} charge into Ba^{137} injections rises to 0.03% at 20th injection

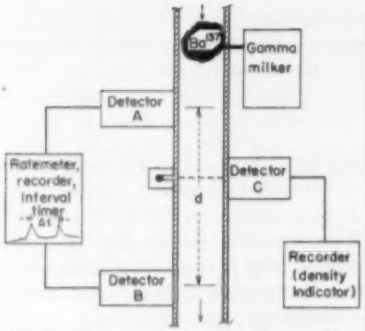

FIG. 5. Setup used to measure mass flow has density gage between two detectors used to find flow rate

Two gamma detectors are spaced a known distance apart on a straight run of pipe. These detectors are connected in parallel and feed a single ratemeter and recorder. As the Ba^{137} passes detector A, an increase in count rate is observed. When the radioactive Ba^{137} passes detector B, a second increase in count rate is observed. The distance between the detectors divided by the time between pulses gives the velocity of the fluid.

Several refinements of this flow system are possible. For example, the mass flow rate can be measured as shown in Fig. 5 by placing a density gage on the pipe. By proper calibration, the system output directly indicates the flow rate in pounds per hour. In these cases a simple computer can be used to combine the time-interval measurement with the density measurement and present a final pounds-per-hour reading.

This method was used in one application of the gamma milker for measuring the rate of flow of a scouring sand in a fluidized-solids system. The distance between detectors was limited to 30 in., for this was the longest run of straight pipe without any restrictions. The range of velocities encountered caused a variation from 3 to 40 sec for the 30-in. run. The density of the flowing solid varied with the velocity.

Other applications. The milker has had several applications in the laboratory for demonstration purposes. The Ba^{137} activity has been used to obtain the coincidence correction and the coincidence-decay curve for a particular counting setup. It has also been used for laboratory half-life experiments.

* * *

The authors gratefully acknowledge the skilled assistance of J. Paul Peterson in planning and carrying out many of the experiments involved in this work.

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BOOKS RECEIVED

American Institute of Physics Handbook, Dwight E. Gray, coordinating editor (McGraw-Hill Book Co., New York, 1957, ~1,525 pages, $15.00). Much of the data of physics that is currently hard to find is now available in the long-awaited AIP handbook: molecular energy levels, gamma-ray energies, the formulas for TE_{011} modes, etc. The field of physics is divided into eight sections, and the information judged most important is given in the form of definitions, nomenclature, laws, formulas, and vast tables of physical constants.

It is unfair, in fact impossible, to evaluate such a volume except by finding out how many times it satisfies and how many times it frustrates in a lifetime. However, it appears at first glance that a more vigorous and more rigorous editorial plan would be an improvement. Some sections end with a bibliography; some start with a bibliography; some have none. Peltier effect is defined on page 5-4 and again on 5-98, but the reader who looks for it will find only the latter in the index. Half-life is not defined or indexed. We can hope that such random sampling is not representative and that with use and growth the book will fill a need beside our venerable "Handbook of Physics and Chemistry."—RHE

Principles of Nuclear Reactors, by Stephen Malaker (International Correspondence Schools, Scranton, Pa., 95 pages). This is a new book in nuclear engineering that has no equations. It is designed for the final course in the ICS nuclear program. This little volume will serve as a valuable introduction to the concepts of nuclear reactors, something that is much needed by those who would like to go on to meatier literature and by those who want a speaking acquaintance to be more effective technicians and administrators.—RHE

Radioisotopes, A New Tool for Industry, by Sidney Jefferson (George Newnes, Ltd., London, 1957, 110 pages, $2.45). A layman's introduction to its subject, this book devotes its first half to industrial applications and its second to the fundamentals of radioactivity. The language is clear and simple.

Electronic Components Handbook, edited by Keith Henney and Craig Walsh (McGraw-Hill Book Co., Inc., New York, 1957, vii + 244 pages, $9.00). This book lists environmental effects on resistors, capacitors, relays, and switches. Military specifications are given for each.

Automation: Its Purpose and Future, by Magnus Pyke (Philosophical Library, Inc., New York, 1957, 191 pages, $10.00). In an interesting, narrative style, the author starts with a description of computers, devotes the major part of the book to applications of automation, and concludes with speculation on automation's impact on society.

Rutherford—Atom Pioneer, by John Rowland (Philosophical Library, Inc., New York, 1957, 160 pages, $4.75). This biography is written with the belief that Rutherford played a major part in shaping the course of modern physics. It follows his life and career without attempting to probe the technical implications of his work.

Inventors and Inventions, by C. D. Tuska (McGraw-Hill Book Co., Inc., New York, 1957, x + 174 pages, $3.75). This book tells how to protect, patent, and market inventions. Case histories suggest how best to use creative abilities.

Tubes for Computers, by Philips Electron Tube Div. (N. V. Philips' Gloeilampenfabrieken, Eindhoven, Holland, 1957, ix + 51 pages, $1.50). This book, no. XII in the Philips' Technical Library, describes special tubes for high-speed and low-speed computers and gives some typical circuits.

U.H.F. Tubes for Communication and Measuring Equipment, by Philips Electron Tube Div. (N. V. Philips' Gloeilampenfabrieken, Eindhoven, Holland, 1957, ix + 60 pages, $1.50). Tube range for U.H.F. and S.H.F. waves is described; noise-factor measurement at these frequencies is also dealt with.

Building an Engineering Career, 3rd ed., by Clement C. Williams and Erich A. Farber (McGraw-Hill Book Co., Inc., New York, 1957, x + 299 pages, $4.75). This guide to the engineering student takes cognizance of engineering developments since the Second World War, e.g., the chapter on "Atomic, Nuclear, and Solar Energy" (the latter heading containing a slightly disturbing redundancy).

Thermodynamics and Statistical Mechanics, by A. H. Wilson (Cambridge University Press, New York, 1957, xv + 495 pages, $9.50). This book, written for the theoretical physicist, avoids much of the mathematical detail as unnecessary for that type of reader. It presents the classical development of thermodynamics, derivation of fundamental statistical-mechanics formulas, and applications.

Nuclear Reactor Physics, by Raymond L. Murray (Prentice-Hall, Inc., Englewood Cliffs, N. J., 1957, xi + 317 pages, $10.00). *(To be reviewed.)*

ALSO OF NOTE

World Wide Industrial Role of Nuclear Energy. In qualitative, easy-to-grasp terms, John Jay Hopkins discusses the role of nuclear energy. This booklet encompasses the impact of nuclear energy on the world economy, the progress being made in various countries, and the elementary principles of nuclear energy. 119 p. *Director of Publications, General Dynamics Corp., 445 Park Ave., New York 22, N. Y., $1.00.*

National Bureau of Standards Annual Report, 1956 (NBS-220). This report summarizes research and development activities in the physical sciences. 158 p. *Superintendent of Documents, U. S. Government Printing Office, Washington 25, D. C., $0.60.*

Magnetic-Amplifier Circuits: Basic Principles, Characteristics, and Applications, 2nd ed., by William A. Geyger (McGraw-Hill Book Co., Inc., New York, 1957, 396 pages, $7.00). This enlarged edition, avoiding extensive mathematics, uses descriptive and graphical methods to interpret essential facts. Material is classified according to circuit functions.

Radiological Health Handbook (PB 121784). Prepared for U. S. Public

Health Service training courses, contains a glossary and sections on physical, chemical, and mathematical data; radioisotopes; decay; radioassay; radiation protection; and a table of isotopes. 356 p. *Office of Technical Services, U. S. Department of Commerce, Washington 25, D. C., $3.75.*

Bibliography of Devices for Damping Mechanical Vibrations (PB 121299). This bibliography, prepared by the Navy, covers literature published between 1924 and Aug. '56. 101 p. *Office of Technical Services, U. S. Department of Commerce, Washington 25, D. C., $2.75.*

Bibliography on Ionizing Radiations, Supplement No. III, Parts I and II. These volumes supplement the series on radiation sterilization of foods prepared by the Quartermaster Food and Container Institute. Part I (PB 111636 S-2)—$4.75; Part II (PB 111636 S-3)—119 p., $3.00. *Office of Technical Services, U. S. Department of Commerce, Washington 25, D. C.*

Contributions on Partially Balanced Incomplete Block Designs with Two Associate Classes. This monograph, by W. H. Clatworthy, is part of National Bureau of Standards Applied Mathematics Series 47. Tables list 75 new designs. 70 p. *Superintendent of Documents, Government Printing Office, Washington 25, D. C., $0.45.*

The Chemical Industry Facts Book. This third edition describes the chemical industry's growth, operation, and technological contributions. 160 p. *Manufacturing Chemists' Assoc., Inc., 1625 Eye St., N. W., Washington 6, D. C., $1.25.*

Encyclopedia of Instrumentation for Industrial Hygiene, edited by C. D. Yaffe, A. D. Hosey, and D. H. Byers (University of Michigan, Institute of Industrial Health and School of Public Health, 1956, xvii + 1,243 pages, $30.00). This volume covers 500 instruments of 60 manufacturers for measuring air contaminants, air velocity and metering, sound and vibration, ionizing radiation and u-v, visible, and infrared radiation.

Meetings of the Academy of Sciences of the USSR on the Peaceful Use of Atomic Energy—July 1 to 5, 1955; Vol. 1, Plenary Meetings. This is the first of five volumes translated from Russian into German. 85 p. *Akademie Verlag, Mohrenstrasse 39, Berlin W8, Germany, $1.57.*

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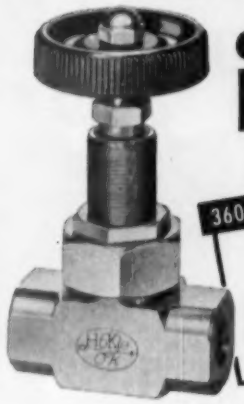

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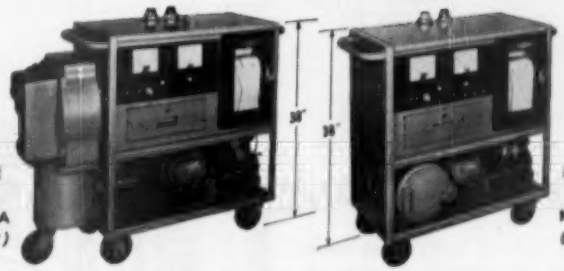

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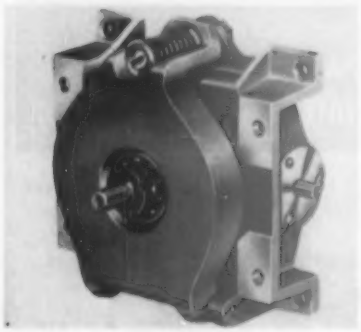
puting analog-to-digital-converter (shown) processes incoming information from thermocouples, pressure gages, steam-analysis instruments.—Beckman Instruments, Inc., 2500 Fullerton Rd., Fullerton, Calif.

IBM-709 Computer

New computer, model 709, has features including independent input-output control, automatic indexing facilities, array of logical operations. Magnetic-core storage has capacity up to 32,768 words or 327,000 decimal digits. Arithmetic and control system has parallel binary design with 36-binary-digit word length. Calculating speeds: 24 μsec—rate about 42,000 per second for addition or subtraction; 240 μsec—rate about 4,000 for multiplication or division; 24 μsec—rate about 42,000 per second for logical operations.—International Business Machines Corp., 590 Madison Ave., New York 22, N. Y.

Computers and Components



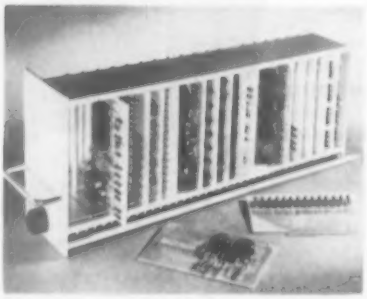
Integrators

Ball-and-disk integrators are used in computing systems requiring mechanical solution of differential equations. Integrators are included in standard electromechanical bread-board parts line.—Helipot Corp., Newport Beach, Calif.

Read-out. Electronic read-out tube converts electronic signals to readable characters. Contains numeric digits, triggered by ¼-watt voltage source.—Electronic Tube Division, Burroughs Corp., Plainfield, N. J.

Translater

Printer-plotter translates output of computer into printed records; reads out information at 300 plotting points per sec. Machine is designed as part of automatic data-handling system.—Burroughs Corp., 6071 Second Ave., Detroit 32, Mich.



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Products Index

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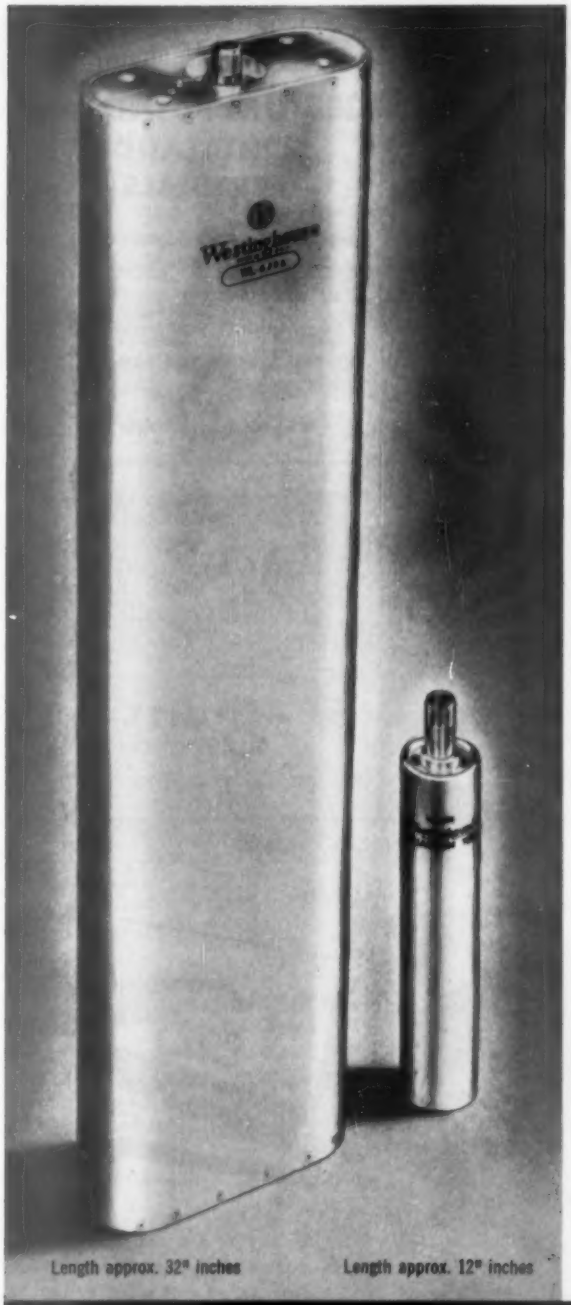

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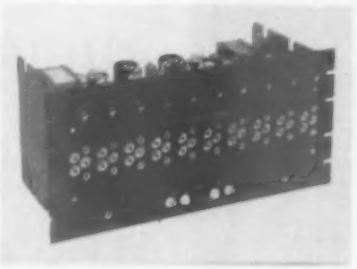
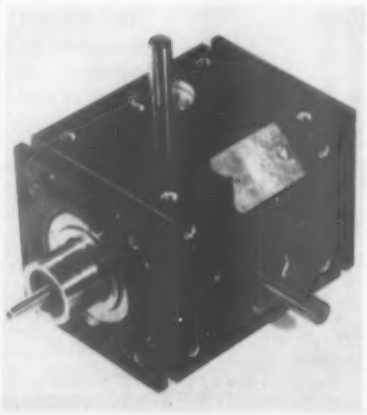
large capacitor banks. Over-all dimensions are: 19¼ × 15 × 12¾ in.—Perkin Engineering Corp., 345 Kansas St., El Segundo, Calif.



Self-Powered Analog

Self-powered operational manifold for use with plug-in computing modules, meters, recorders, oscilloscopes. Triangular arrangement locates amplifier input, output terminals. D-c, a-c operating switches, pilot lamps are mounted on front panel. Power supply offers regulated ±300 volts d-c, 0–10 ma for external use while providing 6.3-volt a-c supply for filaments. Precision voltage reference serves both channels.—George A. Philbrick Researches Inc., 230 Congress St., Boston 10, Mass..



Ball Resolver

Miniaturized precision mechanical ball resolver is used in analog computation applications. Input shaft (0.1248-in. diam) can be driven up to 100 rpm. It requires <5 in.-oz at 25° C for output of 1–2 in.-oz at 0.187-in.-diam sine and cosine output shafts. Orientation shafts concentric with input establishes output relation with accuracy of ±0.5% over a 360-deg range. Component is 2⅛ in. long, weighs <¾ lb with

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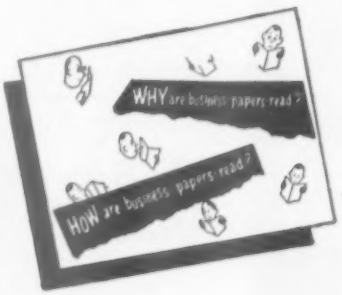

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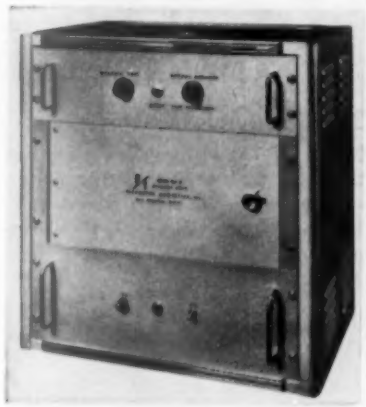

Buffer Storage Unit

Coincident-current magnetic-core storage unit can be used as a temporary store, buffer, or delay unit in data processing, computing and automation systems. Each of 1,092 characters has capacity of seven binary digits in length. Storage unit is completely transistorized.—Telemeter Magnetics, Inc., 2245 Pontius Ave., Los Angeles 64, Calif.

Transistorized Computer

Computer incorporates combination of transitors and printed circuitry. Has initial memory capacity of 4,096 words expandable to 65,536. System consists of central computing unit, paper tape input-output. It can perform 39,000 additions or subtractions per sec, 12,000 multiplications per sec, or 10,000 divisions per sec.—Philco Corp., Philadelphia, Pa.

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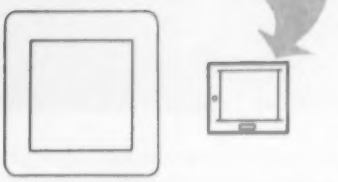

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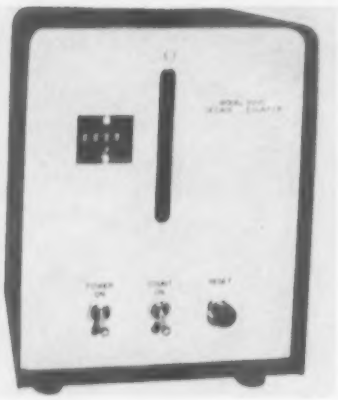


Portable Counter

Seven-pound counter accepts counting rates to 12,000 cpm; combines electronic decade and 5-digit mechanical register. Requires 2.5-volt input.—Performance Measurements Co., 15,301 W. McNichols Rd., Detroit 35, Mich.

Internal-Gas Counters

Designed for measuring tritium and carbon-14 with maximum sensitivity, counters can be used for low-level counting including radiocarbon dating and tritium hydrology. Available in sizes from 100 cm^3 to 5 liters.—Radiation Instrument Development Laboratory, 5737 Halsted St., Chicago 21, Illinois.

Area Monitor

Nucliguard area-personnel-equipment monitor has 3-decade logarithmic count-rate meter. Has built-in calibration sample, visual and audio alarms.—Nuclear Measurements Corp., 2460 N. Arlington Ave., Indianapolis 18, Ind.

Shielded Turntable

Model LS-7M lead shield has 3-tray turntable that permits preparing one sample while another is counted. Has swing-out absorber shelf.—Technical Associates, 140 W. Providencia Ave., Burbank, Cal.

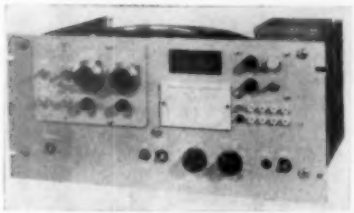
Spectrometer

Model 300 used for alpha, beta, or gamma spectrometry. Has energy-band width ranging from zero to full spectrum. Unit consists of linear pulse amplifier, pulse-height analyzer, precision ratemeter, and high-voltage power supply. Each unit, which has its own power supply, covers 500-2,000 volts and furnishes current up to 1 ma. Regulation is 0.005%.—Mount Sopris Instrument Co., 1320 Pearl St., Boulder, Colo.

Nonoverloading Amplifier

Model is made for scintillation and proportional counters. Amplifier feeds

fast discriminator circuit. Discriminator generates signals of uniform width and amplitude which triggers at 1.5 Mc. Signals sent to external fast scalers or coincidence circuits.—Beva Laboratory, PO Box 478, Trenton, N. J.



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Model 348A combines linear amplifier, single-channel analyzer. Measurements can be made of 662-kev Cs^{137} gammas in Co^{60} background of 1.2×10^7 cpm. Output pulse width is 1.2 μsec; overload recovery, 7 μsec.—Franklin Electronics, E. 4th St., Bridgeport, Pa.

5-Decade Scaler

This scaler has 5 plug-in decades. Single 100,000 indicator permits count to 199,999. Resolution time is 1 μsec. Single lever stops, resets, and starts.—Technical Associates, 140 West Providencia Ave, Burbank, Cal.

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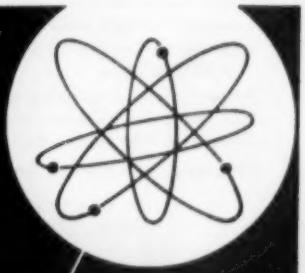

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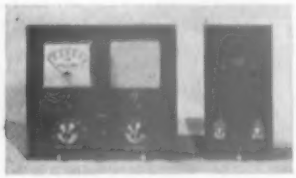

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Film badges. Folder shows body, wrist, and finger film-badge services, also fast-neutron-monitoring badge.—R. S. Landauer, Jr., & Co., 24 Plaza, Park Forest, Ill.

Reactor construction. Bulletin AEB 4-A discusses atomic activities, reactor building.—Babcock & Wilcox Co., 161 E. 42 St., New York 17, N. Y.

Reactor monitoring. Bulletins RM 100–104 discuss four reactor monitoring instruments.—Tracerlab, Inc., 1601 Trapelo Rd., Waltham 54, Mass.

Cobalt-60. Bulletin KC-1 outlines uses of Co^{60} for research and processing.—Atomic Energy of Canada, Ltd., PO Box 93, Ottawa, Canada.

INDUSTRY NOTES

▶**Beryllium Corp.** will convert a railroad roundhouse in Ashmore, Pa. to a Be-extraction plant.

▶**Chem-Pure Service Co.,** San Carlos, Calif., specialists in manufacturing water conditioning and nuclear waste reclamation systems, has doubled its plant facilities.

▶**Universal Atomics Corp.,** Westbury, N. Y. has changed its name to **Universal Transistor Products Corp.**

▶**ACF Industries, Inc.** has combined its **Nuclear Energy Products** and **Erco** (Engineering-Research Co.) divisions. **Marshall G. Holloway** was named president of the combined division whose headquarters is in Riverdale, Md.

▶**Nuclear Instrument Division,** Albuquerque, N. M. is a new subsidiary of **Gulton Industries, Inc.** Nuclear Instrument will specialize in instrumentation for communication links and readout meters.

NEWSMAKERS

Eger V. Murphree, president of Esso Research & Engineering Co., has been named to AEC's General Advisory Committee, succeeding **Eugene P. Wigner,** resigned. The term runs until Aug. 1, 1958.

Samuel Storchheim is president and technical director of Metals Research & Development Inc., a newly-formed firm in Exeter, Pa., to act as a laboratory and consulting service in the fields of metals, cermets and ceramics. It will specialize in reactor core component work.

James H. Gardner and **Arthur G. Walsh** have been promoted to assistant directors, research division at National Research Corp.

Oscar S. Gray has been elected secretary and treasurer of Nuclear Materials & Equipment Co., Apollo, Pa. Formerly a member of the Legal Advisor's Office in the Department of State, he succeeds **Frederick Forscher** and **Leonard Pepkowitz** who had filled the two posts separately; they remain as vice-presidents in charge of metallurgy and chemistry respectively.

AEC has promoted **John W. Ruch** from deputy director to director of the feed materials division at Oak Ridge, succeeding **Francis R. Dowling,** transferred to Washington (NU, April '57, 132). **Berwyn M. Robinson,** deputy director of the Fernald area office, succeeds Ruch.

William W. Lowe, nuclear engineer formerly with Hanford, Los Alamos and Bath Iron Works, Bath, Me., has become a partner in Pickard-Warren-Lowe Associates, Washington, D. C.

Lieuwe J. Dijkstra from the U. of Amsterdam, with Westinghouse since 1953, and **George L. Guthrie,** research engineer in the physics department of Ford Motor Co.'s research labs, have joined the staff of the Hopkins Laboratory of General Dynamics Corp.'s General Atomic division. Both are physicists specializing in properties of metals. General Atomic also named **Harry L. Browne** assistant to the division general manager. Formerly manager, nuclear products department, at Thompson Products, he had earlier been an engineer and administrator at AEC and the Manhattan District.

Phillips Petroleum's atomic energy division has engaged **Bernard Keisch**

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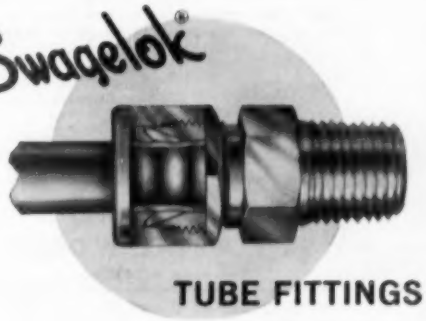

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(IN THE NOVEMBER ISSUE)

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as a separations research chemist, **John H. Ronsick** as a reactor engineer, and **James McKenna** as a reactor project engineer. Joining the Materials Testing Reactor staff are **Kenneth V. Moore** as an irradiations experiment engineer and **Harry L. Kirkham** as a health physicist; at the Idaho Chemical Processing Plant are **Robert Henry**, research chemist, and **Ben M. Hautalla**, health physicist; with SPERT is physicist-engineer **John Grund.**

Graver Tank & Mfg. Co. has elected **E. N. Gosselin,** president since 1930, board chairman. **J. E. Swanson** executive vice-president since 1948, succeeds him as president.

Davis R. Dewey II was named president and director of Baird Atomics, Inc., Cambridge, Mass. **Walter S. Baird,** former president, becomes board chairman. Dewey had been vice-president of High Voltage Engineering Corp.

Niels Bohr was recipient of the first $75,000 Ford Atoms for Peace Award. The award will be granted annually for outstanding contributions to peaceful uses of atomic energy.

William C. Riley joined the ceramic research staff of Battelle Memorial Institute. He had been with General Electric Co.'s Hanford operations.

NUCLEAR CAMPUS

• AEC will sponsor summer institutes in nuclear energy technology for 150 university faculty members at Brookhaven, Argonne, Oak Ridge, Ames lab and Hanford. Inquire of W. L. Collins, Am. Soc. for Engineering Education, U. of Illinois, Urbana.

• One-year AEC fellowships in radiological physics have been awarded 84 students by Oak Ridge Institute of Nuclear Studies. Drawn from 31 states, the fellows will each receive $2,500 plus other allowances. Students will spend the nine months of the '57–'58 academic year at a university, then go to a laboratory for three months' actual experience. Cooperating institutions are Vanderbilt U. with Oak Ridge National Laboratory; Rochester U. with Brookhaven; and the Universities of Washington and Kansas with Hanford Atomic Products Operation.

• 117 other students—from 33 states, the District of Columbia, Canal Zone, Hawaii and Puerto Rico—were awarded AEC fellowships in nuclear technology emphasizing reactor development. The fellowships carry a basis stipend

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New Advertisements
received by May 10th will appear in the June issue, subject to space limitations.
Classified Advertising Division
Nucleonics
P.O. Box 12 New York 36, N. Y.

of $1,800 plus other allowances, for one year's study at one of 28 colleges. This program, like the above, is administered for AEC by ORINS.

• AEC has expanded and systematized its program for giving work experience in AEC laboratories and plants to privately-employed nuclear scientists and engineers. Underway since 1954 at Oak Ridge, the program will now be put on a regular basis. It offers no basic training, but is intended to help persons already trained in nuclear technology to become familiar with specific uses. Applications will be considered on the basis of whether: like training is available outside AEC; the employees are qualified to take full advantage of the work experience; and the assignment places no undue burden on regular work at the AEC facility. Inquire of Director, Division of Civilian Application, AEC, Washington.

Courses

An institute in **theoretical physics** will be held at Brandeis U., June 24–Aug. 2, for graduate students. Write Brandeis U. summer school, Hayden 1-R, Waltham 54, Mass.

The third annual four-week course in **nuclear energy fundamentals for industry** will be held at North Carolina State's engineering school July 1–26. The course includes demonstrations on the college's reactor. Tuition is $500. Write D. B. Stansel, College Extension Division, N. C. State College, Raleigh, N. C.

Three Gordon Research conferences this summer will cover **metals at high temperatures,** June 17–21; **nuclear chemistry,** June 24–28; and **instrumentation,** July 22–26. Write W. G. Parks, U. of Rhode Island, Kingston, R. I.

MIT is offering two-week courses in **nuclear reactor technology,** starting June 24, and **principles of radioisotope utilization,** starting July 8. Tuition is $250. Write Prof. S. H. Crandall, Mechanical Engineering Dept., MIT, Cambridge 39, Mass.

U. of California offers a second annual summer program in nuclear energy for industry, with a nine-week intensive **nuclear engineering short course** July 1–Aug. 30 for engineers needing a basic knowledge in this field, and a one-week **nuclear engineering survey** July 8–12 giving a grounding in atomic energy and its industrial applications to executives who must plan and administer nuclear energy programs.

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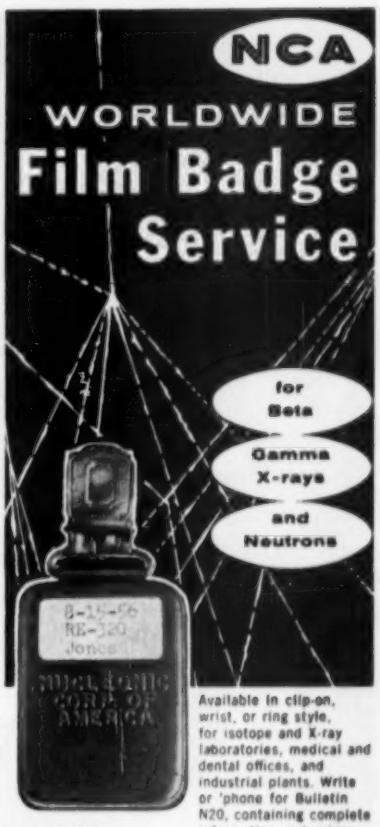

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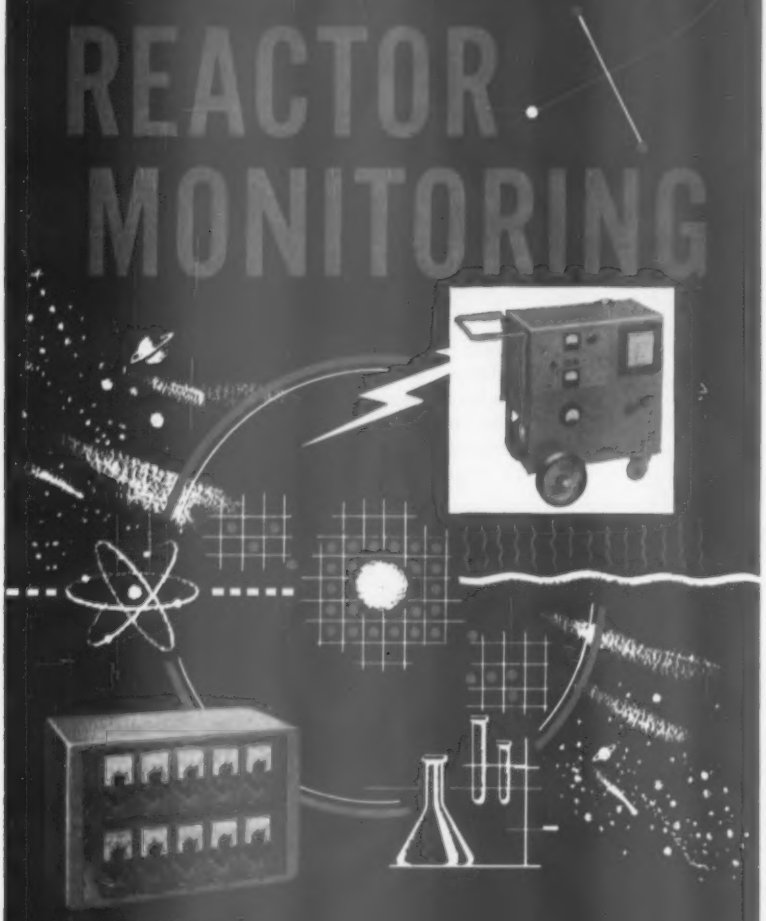

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NUCLEAR CALENDAR

May 13–15—Annual meeting, Radiation Research Society, Rochester, N. Y. Symposia on radiation genetics and on radiation chemistry. Contact A. Edelmann, Box 10901, Pittsburgh 36.

May 14–16—Second annual Industrial Nuclear Technology conference on applied uses of radiation (Armour Research Foundation and NUCLEONICS), Chicago (Museum of Science and Industry). L. Reiffel, ARF, 3440 S. State St., Chicago.

May 28–29—Second reactive metals conference (So. California section AIME), Los Angeles (*Ambassador*). Contact B. S. Mesick, Arthur D. Little Inc., Los Angeles.

June 6–8—Third conference on nuclear structure (Univ. Pittsburgh), Pittsburgh. Contact Norman Austern, Sarah M. Scaife Radiation Laboratory, Univ. Pittsburgh.

June 10–12—Third annual meeting, American Nuclear Society, Pittsburgh (*Wm. Penn-Sheraton*). Contact W. W. Grigorieff, exec. secy., Box 963, Oak Ridge, Tenn.

June 11–15—Third International Conference on Ionization Phenomena in Gases, Italian Physical Society, Milan. One of four sections will be on nuclear reactions in discharges, neutron production, contraction mechanisms, etc.; English is preferred language. Contact Ugo Facchini, CISE, Via Procaccini 1, Milano, Italy.

June 15–Sept. 8—International exhibition "The Atom," near Schiphol airport, Amsterdam. Ten countries including US and USSR will participate.

June 16–21—Annual meeting, Am. Soc. for Testing Materials, Atlantic City (*Chalfonte-Haddon*). Contact Fred F. Van Atta, 1916 Race St., Phila. 3, Pa.

June 17–19—Third annual meeting, Health Physics Society, Pittsburgh (*Pittsburgh U.*). Contact L. C. Emerson, ORNL, Oak Ridge, Tenn.

June 20–23—Fourth annual meeting, Society of Nuclear Medicine, Oklahoma City (*Skirvin*). Contact R. W. Lackey, 452 Metropolitan Bldg., Denver, Colo.

July 1–6—First international colloquium on corpuscular photography, under auspices French National Center of Scientific Research, Strasbourg, France. Aims: to determine state of present knowledge on mechanism of photographic effect of charged particles, and to get a detailed report on Russian work in the field. Contact P. Cuer, director, Corpuscular Physics Laboratory, University of Strasbourg.

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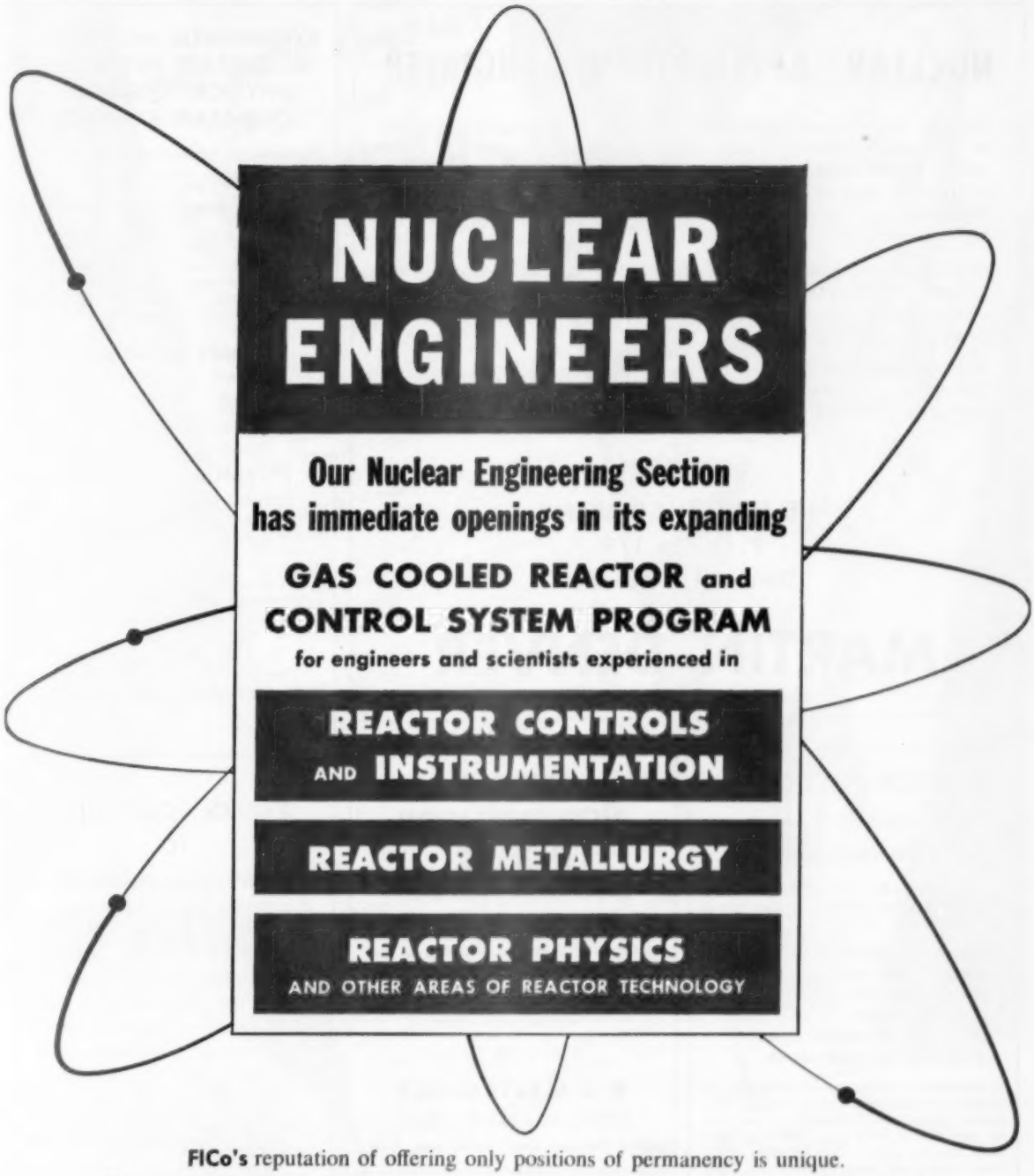

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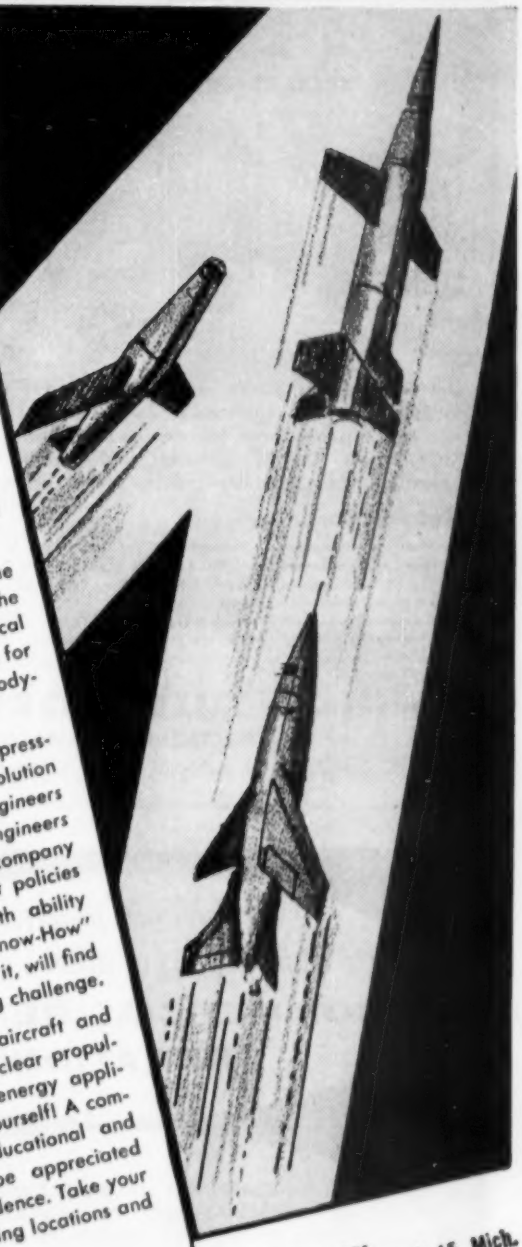

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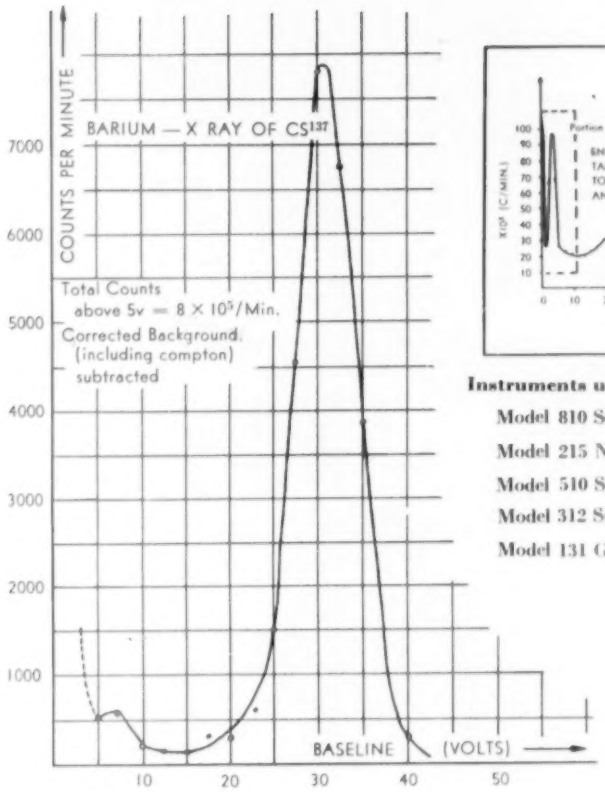

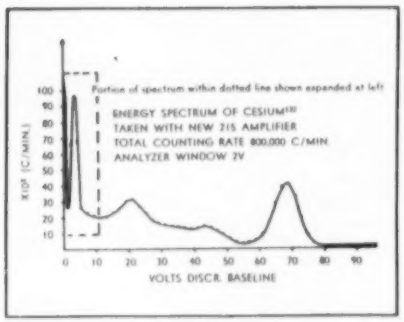
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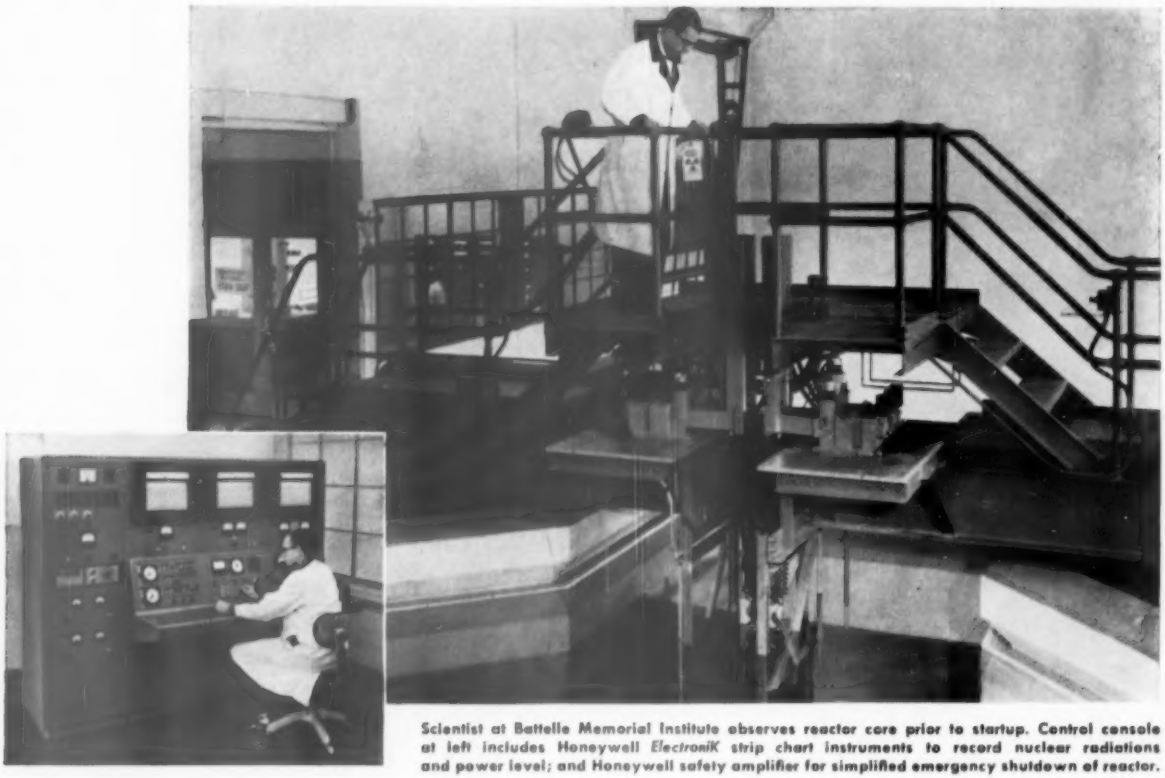

Scientist at Battelle Memorial Institute observes reactor core prior to startup. Control console at left includes Honeywell *ElectroniK* strip chart instruments to record nuclear radiations and power level; and Honeywell safety amplifier for simplified emergency shutdown of reactor.

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